Argonne
NATIONAL LABORATORY

Regulatory Technology Development Plan Sodium Fast Reactor

Mechanistic Source Term Development

Nuclear Engineering Division

About Argonne National Laboratory

Argonne is a U.S. Department of Energy laboratory managed by UChicago Argonne, LLC
under contract DE-AC02-06CH11357. The Laboratory's main facility is outside Chicago,
at 9700 South Cass Avenue, Argonne, Illinois 60439. For information about Argonne
and its pioneering science and technology programs, see www.anl.gov.

Regulatory Technology Development Plan
Sodium Fast Reactor
Mechanistic Source Term Development

Prepared by
David Grabaskas, Acacia Brunett, Matthew Bucknor, James Sienicki, Tanju Sofu
Nuclear Engineering Division, Argonne National Laboratory

February 28, 2015

Acknowledgements

Executive Summary

Construction and operation of a nuclear power installation in the U.S. requires licensing by the U.S. Nuclear Regulatory Commission (NRC). A vital part of this licensing process and integrated safety assessment entails the analysis of a source term (or source *terms*) that represents the release of radionuclides during normal operation and accident sequences. Historically, nuclear plant source term analyses have utilized deterministic, bounding assessments of the radionuclides released to the environment. Significant advancements in technical capabilities and the knowledge state have enabled the development of more realistic analyses such that a mechanistic source term (MST) assessment is now expected to be a requirement of advanced reactor licensing.

This report focuses on the state of development of an MST for a sodium fast reactor (SFR), with the intent of aiding in the process of MST definition by qualitatively identifying and characterizing the major sources and transport processes of radionuclides. Due to common design characteristics among current U.S. SFR vendor designs, a metal-fuel, pool-type SFR has been selected as the reference design for this work, with all phenomenological discussions geared toward this specific reactor configuration. This works also aims to identify the key gaps and uncertainties in the current knowledge state that must be addressed for SFR MST development. It is anticipated that this knowledge state assessment can enable the coordination of technology and analysis tool development discussions such that any knowledge gaps may be addressed.

Sources of radionuclides considered in this report include releases originating both in-vessel and ex-vessel, including in-core fuel, primary sodium and cover gas cleanup systems, and spent fuel movement and handling. Transport phenomena affecting various release groups are identified and qualitatively discussed, including fuel pin and primary coolant retention, and behavior in the cover gas and containment. Radionuclides released from a primary sodium fire are also considered as potential sources. Any available experimental data and pertinent results relevant to the aforementioned phenomena are discussed, and operating incidents at domestically operated facilities are also examined.

Considering the extensive range of phenomena affecting the release of radionuclides, the existing state of knowledge generally appears to be substantial, and may be sufficient in most areas. For core damage accidents, high retention rates should be expected within the fuel matrix and primary sodium coolant for all radionuclides other than the noble gases. These factors greatly reduce the magnitude of possible radionuclide release to the environment.

Several possible gaps within the knowledgebase were identified during this effort. First, there are uncertainties with regard to radionuclide release from metal fuel in the molten state. Another knowledge gap appears in the available thermodynamic data regarding the behavior of lanthanides and actinides in liquid sodium. While not necessarily a phenomenological knowledge gap, a determination of the data requirements for MST development should be formally made prior to the expenditure of significant future research efforts. That is, if additional experimentation is performed in support of MST development, it is important to identify the proper quality assurance requirements for licensing.

Table of Contents

Acronyms

ABTR	Advanced Burner Test Reactor
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ALMR	Advanced Liquid Metal Reactor
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
BWR	Boiling Water Reactor
CRBR	Clinch River Breeder Reactor
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor II
ECN	Energy Research Center of the Netherlands
EPAct	Energy Policy Act
FFTF	Fast Flux Test Facility
FQ	Fuel Qualification
GE	General Electric
HCDA	Hypothetical Core Disruptive Accident
HLR	High-Level Requirements
IFR	Integral Fast Reactor
INL	Idaho National Laboratory
LBE	Licensing Basis Event
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MHTGR	Modular High-Temperature Gas-Cooled Reactor
MST	Mechanistic Source Term
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PRISM	Power Reactor Innovative Small Module
PSAR	Preliminary Safety Analysis Report
PSER	Preliminary Safety Evaluation Report
PSID	Preliminary Safety Information Document
PWR	Pressurized Water Reactor
QA	Quality Assurance
RBCB	Run Beyond Cladding Breach
REVOLS	Release of Volatile Species
RG	Regulatory Guide
RTDP	Regulatory Technology Development Plan
SAFR	Sodium Advanced Fast Reactor
SNL	Sandia National Laboratory

SER   Safety Evaluation Report
SFR   Sodium Fast Reactor
SRE   Sodium Reactor Experiment
SRM   Staff Requirements Memorandum
SRP   Standard Review Plan
TID   Technical Information Document
TRACER Transport Phenomena of Radionuclides for Accident Consequence Evaluation of Reactor
TREAT   Transient Reactor Test Facility
TWR   Traveling Wave Reactor

1 Introduction

To license and operate a commercial nuclear power plant, it is vital to ensure the safety of the public and environment by providing protections against the release of radionuclides. As part of the process to assess the safety of a nuclear plant design, the source term expected to occur during normal reactor operation and accident sequences is analyzed, where a source term is considered to be the types and amounts of radionuclides that could be released to the environment. Not only is this a fundamental part of the design process, but it is a requirement for regulatory licensing [1]. While past nuclear plant power source term analyses often used a deterministic, bounding, and conservative assessment of radionuclide release, over the past 25 years there has been an increased interest in the development of more realistic evaluations that also consider important uncertainties.

This work focuses on the state of development of such a source term, referred to as a mechanistic source term (MST), for sodium fast reactors (SFRs). In general, an MST attempts to realistically model the release and transport of radionuclides from the source to the environment for a specific scenario, while accounting for retention or transmutation phenomena and any associated uncertainties. Determining an MST for radionuclide transport that involves complex phenomena requires significant knowledge and modeling capabilities. Assessing the processes that affect radionuclide transport in sophisticated systems can be difficult, as multiple chemical and physical interactions are occurring simultaneously. Yet, development of a technically sound assessment of the source term is a responsibility of reactor vendors who are seeking to design, license, and build commercial nuclear power plants within the U.S.

To assist in the process of MST formulation, this work seeks to qualitatively identify, describe, and examine the sources and transport processes of radioactive materials in a metal-fuel, pool-type SFR, which is characteristic of current industry designs. This effort also seeks to collect pertinent information and identify key gaps in the current state of knowledge that may need addressed if SFR technology is to become a viable component in the domestic commercial energy market. These tasks are seen as necessary first steps to facilitate subsequent technology and analysis tool development as part of the overall MST development process.

This work, outlined in the following figure, begins with a review of previous source term analyses and SFR development in the U.S., providing background and context for the current task. Following this, sources of radionuclides within a conceptual metal-fuel pool-type SFR are identified and reviewed. The subsequent two sections examine the radionuclide transport phenomena associated with each source, which includes an overview of the current knowledge state and past analyses. Lastly, an assessment of current capabilities is presented, with a description of gaps in the current state of knowledge. Summaries at the beginning of longer sections provide an overview of the key points and essential insights.

Section 2: Background

- Scope and Objectives of Current Work
- Source Term History
- Sodium Fast Reactor Source Term Experience
- Vendor Designs and Reference Design

Section 3: SFR Sources of Radioactivity

- Fuel
  In Core
  Spent Fuel Movement
- Ex-Fuel
  Primary Sodium
  Cover Gas

Section 4: Core Damage Source Term Phenomena

- Transport Phenomena
- State of Knowledge
- Past Accidents & Experiments

Section 5: Ex-Core Source Term Phenomena

- Primary Sodium Fire
- Spent Fuel Movement Accident
- Primary Sodium Cleanup System Accident
- Cover Gas Cleanup System Accident

Section 6: Assessment of Tools and Gaps

- Modeling Capabilities
- State of Knowledge Gaps

2 Background

This section begins by defining an SFR MST and the associated development process, which is utilized to outline the objectives and scope of the current work. The second subsection provides perspective on the current task by reviewing the history of regulatory source term analyses for U.S. nuclear power plants. The third subsection discusses past SFR licensing experience and source term strategies. Finally, current SFR vendor designs are reviewed and a base SFR design is identified, which is provided as a common frame of reference for subsequent sections of the report.

2.1 Objectives and Scope of Current Work

> **Objective and Scope Summary**
> An SFR MST is the result of an analysis of radionuclide release, in terms of quantities, timing, and other characteristics, resulting from a specific event sequence. The current task seeks to aid in SFR MST development by identifying and characterizing radionuclide sources and potential transport pathways and phenomena for a metal-fuel pool-type SFR. Pictorial radionuclide transport diagrams and relational data are presented to aid in issue communication, and a preliminary assessment of gaps in the state of knowledge and modeling capabilities is reviewed.

While the reactor source term is an important part of a variety of reactor design licensing review issues, perhaps the most significant role relates to the assessment of reactor siting and the evaluation of the consequences of licensing basis events (LBEs)[1]. In U.S. regulation, the source term is defined in 10 CFR § 50.2 [2] as:

> The magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

As will be described in the following sections, this source term definition has been used in the establishment of deterministic accident source terms for the operating light-water reactor (LWR) fleet in satisfaction of the requirements of 10 CFR § 50.67 [1]. However, beginning in the early 1990s, regulatory interest increased in the use of more realistic source term evaluations for advanced reactor designs. While there is no formal definition of an MST within U.S. regulation, in 1993, the NRC staff described an MST in SECY-93-092 [3] as:

> …the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

---

[1] Licensing basis events (LBEs) typically represent a range of events expected to occur from normal operation to rare, off-normal conditions (such as anticipated operational occurrences, design basis events, and beyond design basis events).

Given that the focus of the current work is on SFRs, this NRC MST description and the source term definition from 10 CFR § 50.2 can be leveraged to create an SFR MST definition, which is used to establish the scope of the current work:

> An SFR MST is the result of an analysis of radionuclide release, in terms of quantities, timing, and other characteristics, resulting from the specific event sequence being evaluated. It is developed using best-estimate phenomenological models of the transport of radionuclides from the source through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

As can be seen, there are two key differences between this SFR MST definition and the NRC description given in SECY-93-092:

1) The broader term *radionuclide* is used rather than *fission products*, as not all radionuclides present in the SFR system are a direct result of fission.

2) All significant radionuclide sources are included, rather than only radionuclide released from core damage accidents. Due to a desire to reduce the probability and severity of SFR core damage accidents, the ex-core sources of radionuclides, which are discussed in Section 3, may become increasingly important to the MST assessment.

The process for developing and applying an MST is outlined in Figure 2–1. This activity begins with the characterization of radionuclide sources within the reactor system, including the identification of possible knowledge gaps and uncertainties. Next, the potential pathways of radionuclide transport, including associated phenomena, must be characterized. Again, gaps and uncertainties should be identified. From there, efforts to model the phenomena and pathways can begin. These models are then used to evaluate the scenarios of interest, and the results are presented to the regulator for review. Not shown in the figure are several feedback loops, as gaps and uncertainties discovered throughout the process should be used to inform and revise other steps in the MST development.
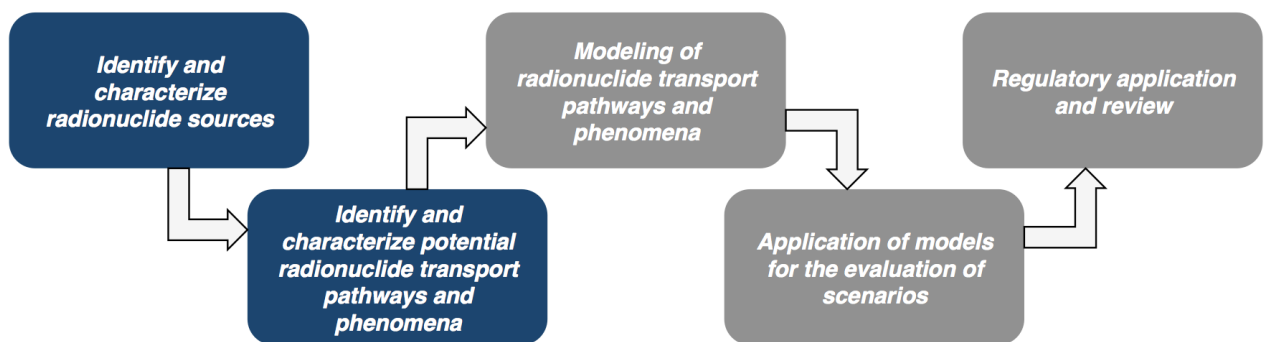
Figure 2–1: MST Development Pathway

The objective of the current work is to address the first two steps of the MST development process: the characterization of radionuclide sources and potential transport pathways and

phenomena. Early development of the methods, data, and analytical tools necessary for design-dependent MSTs has the potential to reduce uncertainty and expedite licensing efforts for advanced reactor vendors as they complete the subsequent steps in MST development. The results of interchanges between SFR vendors and research/development organizations, augmented by summary relational information regarding the relevant phenomena associated with SFR radionuclide transport, can be used to support future interactions with the NRC as part of the SFR pre-application and license review process.

As part of this work, plausible factors that potentially affect an MST analysis for an SFR are identified via qualitative relational models. Because all current U.S. SFR designs utilize a metal-alloy fuel in a pool-type design (as described in Section 2.4), and in an attempt to narrow the scope of this effort, the relational model and contributing analyses will focus on phenomena prevalent in a metal-fuel, pool-type SFR.

The scope of this work can be more formally divided into four central tasks:

1. *Identify sources of radionuclides in a generic metal-fuel, pool-type SFR, and identify the potential inventory of radionuclides associated with each source.*

   The inherent safety and defense-in-depth features of SFRs could preclude core damage events as the largest source of radionuclide released to the environment. Instead, an MST may be derived from accidental releases during refueling/spent fuel handling, sodium fires, or radionuclides in primary and cover gas cleanup systems that, while small, may be larger than credible core damage accident releases.

2. *Identify relevant phenomena associated with radionuclide transport from the sources of radionuclides to the environment.*

   The phenomenological behavior of radionuclide retention and transport can vary significantly depending on the origin location and source. In some cases, defense-in-depth barriers have the potential to reduce the magnitude of the source term, while other phenomena may affect the species and composition of the source term.

3. *Create an intuitive pictorial diagram outlining the relevant transport phenomena for the sources of radionuclides considered.*

   A diagram[2] will be created for each source of radionuclides. The relational diagram is intended to provide an overview of the important phenomena affecting an MST and highlight sources and their corresponding barriers.

4. *Provide a preliminary assessment of current capabilities to evaluate the relevant phenomena.*

---

[2] Similar to the MST diagram created for the NGNP project [11].

There has been substantial previous effort dedicated to understanding radionuclide behavior during SFR accidents, and a myriad of tools are currently available for assessment of some prominent MST phenomena. A preliminary assessment of the adequacy of the current state of knowledge and modeling capabilities will be performed, and gaps in the knowledge state or capabilities will be identified.

While the focus here is on SFR MST development for reactor siting and the evaluation of LBEs, similar analyses could be used for the satisfaction of other regulatory requirements, such as the design requirements relating to control room ventilation and emergency system shielding.

2.2 Source Term History

2.2.1 Current Regulatory Requirements

Simply stated, the purpose of the regulator is to ensure the public is protected from any undue risks created by a nuclear power plant. There is a direct relation between this regulatory responsibility and source term analyses, which is why it is essential to the licensing process and has been a focal point of previous licensing efforts.

Due to the historical prevalence of LWRs in the U.S. commercial reactor fleet, the current regulatory environment is largely focused on the analysis and regulation of LWR designs. A component of the licensing process includes the analysis of potential radiological consequences to the public based on an NRC-prescribed source term. Initial source term parameters for LWRs were developed in the infancy of the commercial industry using knowledge available at the time. Accordingly, advances in the state-of-the-art have led to revised regulatory source terms. The various iterations of the regulatory requirements as they pertain to source terms are discussed in this section.

TID-14844, An LWR Source Term

> **TID-14844 Summary**
> *TID-14844 defined the source term for a maximum credible accident at an LWR based on the current state of knowledge at the time of its publication in 1962. Assumptions from the TID were ultimately used in formulation of the NRC's formal guidance in Regulatory Guides 1.3 and 1.4. The deterministic analysis used conservative assumptions and release fractions. In general, the TID-14844 source term is unsuitable for SFRs, as the phenomena associated with the base accident are not comparable to SFR accidents.*

In 1962, the U.S. Atomic Energy Commission (AEC) released the Technical Information Document (TID) titled "Calculation of Distance Factors for Power and Test Reactor Sites" [4], also known as TID-14844. This report outlined a deterministic methodology for calculating radiological hazards and corresponding site boundary distances at commercial nuclear power plants. TID-14844 included guidance regarding the assumed fractional release to containment, atmospheric transport and dispersion behavior, and calculation of offsite consequences. At this point, it is important to stress that TID-14844 only contained explicit guidance for LWRs, and in the absence of adequate experimental and operational data, utilized conservative parameters and assumptions, such as the assumption that the source term is released instantaneously into containment.

The source term cited in TID-14844 was based on deterministic assumptions for a maximum credible accident in an LWR, which was loosely defined in the TID as a substantial core melt resulting from a loss of coolant accident (LOCA). The assumptions of the simplified source term have been incorporated into the NRC's Regulatory Guides (R.G. 1.3, R.G. 1.4) for evaluation of radiological consequences at power reactors. These release fractions, shown in Table 2-1 with other key assumptions of the source term, were derived from experiments in the 1950s performed on irradiated uranium oxide pellets, but are not explicitly correlated with phenomenological behavior.

Table 2-1: Key Assumptions of TID-14844 Source Term

Parameter	Assumptions
Accident sequence	Maximum credible accident for LWR: pipe rupture resulting in massive core melt
Release to containment	Noble gases: 100% Halogens: 50% Remaining solids: 1%
Containment leakage rate	Maximum allowable: 0.1% per day
Engineered safety features	No credit
Release period	Instantaneous

In general, the TID-14844 source term is unsuitable for use in establishing a credible SFR source term due to key differences in associated accident phenomena. In TID-14844, the maximum credible accident of a large break LOCA could create a direct pathway from the core, as it is uncovered due to the loss of coolant, to the containment through the piping break. This is fundamentally different from SFR accidents, where LOCAs are generally considered to be of extremely low probability (especially for pool-type designs), and the creation of a direct pathway for radionuclides from the core to containment is unlikely, as will be discussed in Section 4. Additionally, unrealistic layers of conservatism are introduced via the key assumptions of TID-14844, such as *instantaneous* release to containment without time consideration of retention or holdup phenomena.

NUREG-1465, The Revised LWR Source Term

> **NUREG-1465 Summary**
> NUREG-1465 addressed the limitations of TID-14844 by developing unique releases for BWRs and PWRs based on accident scenarios from NUREG-1150 and supplemental analyses. NUREG-1465 also includes timed-releases with credit for engineered safety features along with uncertainty analyses. As with TID-14844, the release fractions of NUREG-1465 are considered unsuitable for SFRs.

More than thirty years after the publication of TID-14844, the NRC released a revised accident source term in NUREG-1465 [5]. This NUREG attempted to address the shortcomings that resulted from the conservative, simplistic assumptions of TID-14844. NUREG-1465 presents unique boiling water reactor (BWR) and pressurized water reactor (PWR) source terms that are based on a range of accident scenarios derived from NUREG-1150 analyses [6]. NUREG-1465 explicitly addresses fuel failure phenomena, quantitatively considers uncertainties, and provides guidance on in-containment retention mechanisms.

Experimental efforts since the release of TID-14844 led to the understanding that releases in an LWR can be categorized into several phenomenological phases, where the release to the environment is directly affected by the severity of each phase. Because it was recognized that the guidance provided in TID-14844 was not realistic (*i.e.*, very conservative) and could inadvertently lead to decisions that negatively impacted safety, the primary intent of NUREG-1465 was to act as a basis for revisions to regulatory requirements for existing and future LWRs. Beyond the explicit consideration of unique BWR and PWR source terms for a range of accident sequences, NUREG-1465 differed from TID-14844 in that it attempted to provide a realistic

assessment (including consideration of uncertainties) of the source term, rather than a deterministic bounding value. NUREG-1465 also allowed credit for fission product removal via the use of engineered safety features, which TID-14844 did not consider, and it included a finer discretization of release groups.

In NUREG-1465, all experimental data were gathered using low burnup fuel; however, it is currently understood that the burnup level may affect the final failure threshold and timing and morphology behavior following failure. The uncertainties recognized in NUREG-1465 that are expected to affect the magnitude of release include the accident type/severity, time of onset of release, durations of each release phase, composition/magnitude of releases, and the chemical form of iodine[3].

The revised source term is shown in Table 2-2. Key assumptions used in the development of the source term are listed in Table 2-3. While the intention of NUREG-1465 was to identify a scenario that was representative of a range of credible accidents, a large break LOCA was selected because it was expected to result in the earliest onset of radionuclide release to the fuel-clad gap (*i.e.*, gap release). It is important to stress however that the releases were intended to be representative or typical of those resulting from a core melt event at low pressure, and are not necessarily bounding, as was the intent of TID-14844.

Table 2-2: Fractional Release of Core Inventory from NUREG-1465

		Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
BWR	**Duration (hr)**	**0.5**	**1.5**	**3.0**	**10.0**
	Noble gases	0.05	0.95	0	0
	Halogens	0.05	0.25	0.30	0.01
	Alkali metals	0.05	0.20	0.35	0.01
	Tellurium group	0	0.05	0.25	0.005
	Barium, Strontium	0	0.02	0.1	0
	Noble metals	0	0.0025	0.0025	0
	Cerium group	0	0.0005	0.005	0
	Lanthanides	0	0.0002	0.005	0
PWR	**Duration (hr)**	**0.5**	**1.3**	**2.0**	**10.0**
	Noble gases	0.05	0.95	0	0
	Halogens	0.05	0.35	0.25	0.1
	Alkali metals	0.05	0.25	0.35	0.1
	Tellurium group	0	0.05	0.25	0.005
	Barium, Strontium	0	0.02	0.1	0
	Noble metals	0	0.0025	0.0025	0
	Cerium group	0	0.0005	0.005	0
	Lanthanides	0	0.0002	0.005	0

[3] It is important to emphasize that these are the high impact uncertainties recognized in NUREG-1465. For an SFR, radionuclides other than iodine, and their associated chemistry, may have a significant effect on the uncertainty space.

Table 2-3: Key Assumptions of NUREG-1465 Source Term

Parameter	Assumptions
Accident sequence	A credible accident representative of the average scenario based on sequences considered in NUREG-1150 and supplementary analyses. Because a large break LOCA is expected to produce the earliest fuel failure, it has been identified as the representative scenario
Fractional release to containment	See Table 2-2
Engineered safety features	Provides quantitative guidance for containment sprays, BWR suppression pools, filtration systems, water overlying debris, and aerosol deposition
Release period	Timed-release, See Table 2-2

Fractional releases are presented for both BWR and PWR designs, where the releases are dependent on the phenomenological phase. The gap release phase occurs at the onset of fuel failure and results in the release of radionuclides from the fuel-cladding gap. The early in-vessel release phase covers the period of integrity/geometry degradation prior to reactor vessel failure. Ex-vessel release commences with vessel failure and concurrently proceeds with late in-vessel release. Durations of each phase vary for each reactor design, largely due to the power density in the core region.

Like TID-14844, the NUREG-1465 release fractions are inappropriate for SFRs. The accident sequences considered in NUREG-1465 are focused on LWRs, and the timing and species of radionuclide release are not directly translatable to significantly different reactor designs, such as SFRs.

Regulatory Guide 1.183, The Alternative Source Term for LWRs

> **R.G. 1.183 Summary**
> Based on NUREG-1465, R.G. 1.183 provides guidance on using an alternative (to TID-14844) source term for compliance with the source term requirement of 10 CFR § 50.67. While the release fractions in R.G. 1.183 are not suitable for SFRs as they are derived from the LWR analyses in NUREG-1465, the documented criteria for developing an alternative source term are generic in nature and can provide guidance for SFR siting source terms.

In 2000, the NRC released Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" [7] based on the information provided in NUREG-1465. R.G. 1.183 contains formal guidance regarding development of an acceptable source term and completion of offsite consequence and siting analyses. Information in the Regulatory Guide is intended to be applied by the licensee only in conjunction with an acceptable Alternative Source Term (AST)[4]; otherwise, guidance provided in Regulatory Guides 1.3 and 1.4 (based on TID-14844) should be used when performing siting studies.

---

[4] It should be noted that an AST is a deterministic source term developed for the existing LWR fleet, whereas an MST is, by definition, a mechanistic and scenario dependent assessment of radionuclide release.

Similar to NUREG-1465, the release fractions in R.G. 1.183 are meant to be representative of a range of credible accident scenarios in LWRs, although a large break LOCA is cited as the representative accident. R.G. 1.183 also gives a quantitative assessment of the fractional fission product composition of the gap, which applies to non-LOCA scenarios only. The release groups considered in NUREG-1465 are also considered in R.G. 1.183; however, only the gap release and early in-vessel phases are explicitly considered in R.G. 1.183. This is because the design basis source term criteria in 10 CFR § 50.67 [1] considers the total effective dose equivalent over the worst two hours of the accident, which is assumed to be covered by the first two release phases. Because the assumptions presented in the AST were inconsistent with previous NRC guidance in TID-14844 and the dose limits of 10 CFR § 100.11, revised dose guidelines were released in 10 CFR § 50.67 that are intended to supersede those of 10 CFR § 100.11 when used in conjunction with an AST. Accordingly, Regulatory Guide 1.183 and 10 CFR § 50.67 were released simultaneously by the NRC.

The release fractions per group for each phase as well as the time of onset and duration of each phase are shown in Table 2-4. It is important to note that these release fractions and durations match those reported in NUREG-1465 for each phase, but the onset of each phase is now also explicitly defined. The same key assumptions used in development of the revised source term in NUREG-1465, as shown in Table 2-3, also apply to R.G. 1.183.

Table 2-4: Release Fractions to Containment from Regulatory Guide 1.183

		Gap Release	Early In-Vessel
BWR	Onset (s)	120	1800
	Duration (hr)	0.5	1.5
	Noble gases	0.05	0.95
	Halogens	0.05	0.25
	Alkali metals	0.05	0.20
	Tellurium group	0	0.05
	Barium, Strontium	0	0.02
	Noble metals	0	0.0025
	Cerium group	0	0.0005
	Lanthanides	0	0.0002
PWR	Onset (s)	30	1800
	Duration (hr)	0.5	1.3
	Noble gases	0.05	0.95
	Halogens	0.05	0.35
	Alkali metals	0.05	0.25
	Tellurium group	0	0.05
	Barium, Strontium	0	0.02
	Noble metals	0	0.0025
	Cerium group	0	0.0005
	Lanthanides	0	0.0002

Criteria for development of an acceptable AST supplemental to 10 CFR § 50.67 are also provided in R.G. 1.183. These criteria are as follows [7]:

- The AST must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events

that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.

- The AST must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

- The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

- The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.

- The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the AST.

While the release fractions in R.G. 1.183 are unsuitable for SFRs, the AST criteria for satisfying 10 CFR § 50.67 are largely still applicable[5] as they are generic in nature. These criteria can provide guidance on selecting SFR source term scenarios and insight into the type of mechanisms that could be credited for radionuclide retention.

2.2.2 Regulatory History of MST Development

Regulatory discussions regarding the use of MST analyses for advanced reactor designs began in the early 1990s. This section reviews MST history, including recent NGNP/NRC interactions, and new standard requirements.

NRC's Perspective on Use of MSTs for Advanced Reactor Design Licensing

> **NRC Perspective on MST for Advanced Reactors Summary**
> *The NRC formally acknowledged the need for MST approach development in the early 1990s as advanced reactor vendors (including SFRs) entered into preliminary licensing talks. The regulatory staff and Commission affirmed support for use of an MST in siting studies, as did the ACRS, but acknowledged that sufficient data should exist to provide adequate confidence in the MST approach.*

In the early 1990s, the NRC began formally addressing the use of MSTs in advanced reactor licensing with the issuance of SECY-93-092 [3] following a request from the Commission for a review of the state-of-the-art of source term analyses. The SECY, which provides recommendations to the Commission from the staff on several regulatory issues, reviewed the

---

[5] The first AST criteria of R.G. 1.183 states that the AST must address events that include substantial core meltdown with an appreciable release of fission products. As will be discussed in Section 3 and 4, this criteria may be unnecessarily conservative for metal-fuel pool-type SFRs, as the inherent safety design features may reduce the probability of substantial core melt events below the range analyzed for licensing discussions.

vendor-proposed source terms for advanced reactors currently in the preapplication stage. Of the four advanced reactor designs under consideration, only one design proposed the use of an MST. Ultimately, the staff recommended that [3]:

 "...source terms should be based upon a mechanistic analysis and will be based on the staff's assurance that the provisions of the following three items are met:

- *The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.*
- *The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanisms or barrier are not obscured.*
- *The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties."*

The Advisory Committee on Reactor Safeguards (ACRS) agreed with the staff's recommendation in a letter to Chairman Selin in 1993 [8], but also noted that with regard to use of mechanistic analyses, "…it is clear that the present data base will need to be expanded." The Commission formally approved the staff's recommendation regarding use of an MST in a July 30, 1993 Staff Requirements Memorandum (SRM) [9].

The issue was revisited in SECY-03-0047 [8], with the staff again recommending the use of MSTs, specifically for licensing decisions related to the containment and siting. The staff recognized that development of sufficient analytical tools and databases would be necessary and effort intensive, but that use of an MST instead of a deterministic, bounding source term would allow for considerations of unique advanced reactor design features that may affect the source term.

NGNP/NRC Interactions on an MST

NGNP/NRC MST Interactions Summary
As a result of the EPAct of 2005, the DOE/INL entered into preliminary licensing discussions with the NRC regarding the MHTGR. The majority of the NRC's comments on an MST for the MHTGR involve development of verified analytical tools and data on radionuclide transport that could support an MST. The NRC also stressed that while characteristic accidents are important to MST analyses, bounding scenarios must also be considered. The same qualitative feedback can be expected to apply to any SFR-specific licensing discussions.

The Next Generation Nuclear Plant (NGNP) Project was established by the U.S. Department of Energy (DOE) and Idaho National Laboratory (INL) as part of the Energy Policy Act (EPAct) of 2005. The primary objectives of the ongoing project are to develop, design, build, and operate a prototype Generation-IV facility that could generate electricity and/or hydrogen. Early in the

project, the EPAct of 2005 designated the modular high-temperature gas-cooled reactor (MHTGR), with options of a prismatic block or pebble bed fuel type, as the NGNP.

As part of the EPAct of 2005, it was determined that the NRC would have licensing and regulatory authority for the project, requiring the DOE/INL to complete the formal licensing process, which includes development of an acceptable MST [10]. Subsequently, the DOE/INL issued a white paper describing the anticipated approach for development of an MST [11], where methods are intended to be sufficiently generic such that they apply to both proposed fuel types. Because identification of fuel failure mechanisms and thresholds is part of the fuel qualification (FQ) process, a separate white paper issued by the DOE/INL on FQ [12] addresses some information relevant to proposed MST analyses, such as the quantification and development of empirical correlations for fission product transport and release based on fuel performance experiments.

The DOE/INL stated several outcome objectives in the MST white paper regarding what specific feedback it was seeking from the NRC. These resolution issues include:

- Determination that the definition of event specific MSTs for the MHTGR is acceptable. The DOE/INL proposed that an MST be developed for each LBE.
- Determination that the approach to calculation of event specific MSTs for the MHTGR is acceptable, subject to validation of the design methods and supporting data that form the basis of the calculations, with particular focus on the following:
  - Generation and transport of fission products from the fuel kernel to the coolant.
  - Quantification of each species (concentration and form) in the helium circuit during normal operation and assessment of their transport/deposition in the system.
  - Quantification and release of each species from the helium pressure boundary during a depressurization event.
  - Mechanisms affecting the time-dependent transport of radionuclides in and release from the reactor building.
- Determination that the planned fission product transport tests which are part of the FQ program are adequate to validate analytical tools.

Subsequent to the initial release of the white papers in 2010, the NRC and the DOE/INL were actively engaged in pre-licensing discussions regarding FQ and MST development. In general, the most recent feedback provided by the NRC [13] [14] in response to the specific outcome objectives above, indicated that given the maturity of the design, the NRC found the methods proposed in the white paper to be generally reasonable, albeit with some caveats. With regard to the event specific MSTs, the NRC indicated that bounding events such as those identified in NUREG-1338 [15], should be considered for the MHTGR design in addition to licensing basis events. Generally, the NRC concluded that the DOE/INL's planned testing and research activities for MST development were reasonable approaches for identification and evaluation of key fission transport phenomena and associated uncertainties. The NRC staff also indicated that they anticipated the development of information on release and transport phenomena as the result of the DOE/INL's efforts that could be used to support licensing of the MST.

Based on the current status of licensing discussions between the NRC and the DOE/INL regarding development of an MST for the MHTGR, some broad conclusions can be made relative to the use of the MST in general. The majority of regulatory concerns surround the development and validation of models and supporting data that can treat the phenomenological behavior of fission product generation and transport through the various defense-in-depth barriers. In the absence of a large catalog of operating and experimental data, uncertainty identification and quantification of sources of uncertainty becomes a key component of an MST analysis. Given an appropriate treatment of uncertainties, the expected behavior of fission product generation and transport can be enveloped.

ASME/ANS PRA Standard for Advanced Non-LWRs

> **Standards Summary**
> *The recently issued ASME/ANS PRA standard for advanced, non-LWRs details six objectives of an MST analysis as it would apply to a PRA framework. These objectives, which include identification of radionuclides, release pathways, relevant phenomena and key uncertainties, are sufficiently technology neutral that they could be applied to an SFR MST analysis.*

In late 2013, the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) released the "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants" [16]. The intent of this standard is to define the technical requirements and application methodology for probabilistic risk assessments (PRAs) used in risk-informed decision making for advanced reactor analyses. Given the myriad of existing advanced non-LWR designs, the requirements are written to be technology neutral and cover a wide scope of PRA applications, including the quantification and analysis of MSTs.

The requirements of the standard are divided into 18 elements that would typically comprise a full-scope PRA; analysis of an MST is one of these elements. The standard is structured such that each element is described by qualitative objectives and several high-level requirements (HLRs), with additional supporting requirements for each HLR. The HLRs for an MST analysis as they are listed in the standard are shown in Table 2-5. Six objectives of an MST analysis are listed in the standard. These objectives include:

- Identification of inventories available for release within the reactor coolant system pressure boundary,
- Identification and characterization of the phenomena affecting radionuclide transport,
- Definition of reactor-specific release categories for use in end state and event sequence grouping,
- Determination of release parameters (e.g. chemical phase, release timing and duration, etc.),
- Identification and evaluation of relevant uncertainties, and
- Documentation of the mechanistic analysis.

Table 2-5: High-level Requirements for MST Analysis from ASME/ANS PRA Standard for Non-LWRS [16]

Designator	Requirement
HLR-MS-A	Release categories shall be defined for defining event sequence end states and for grouping event sequences and event sequence families with the same or similar mechanistic source terms.
HLR-MS-B	The mechanistic source term analysis shall include a method for determining the mechanistic source term for each release category.
HLR-MS-C	The mechanistic source term analysis shall include calculations to quantitatively characterize the mechanistic source terms for each release category.
HLR-MS-D	Uncertainties in the mechanistic source terms and associated radionuclide transport phenomena shall be characterized and quantified to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. Those sources of uncertainty that are not quantified shall be addressed via sensitivity analysis.
HLR-MS-E	The mechanistic source term analysis shall be documented consistent with the applicable supporting requirements.

2.3 Sodium Fast Reactor Source Term Experience

While several prototype and proof-of-concept SFRs have been constructed and operated successfully in the U.S., as seen in the figure below, none have pursued the NRC's commercial licensing process. The Fermi 1 fast breeder reactor was licensed by the NRC's predecessor, the Atomic Energy Commission (AEC), but no formal source term analysis was conducted. The second experimental breeder reactor (EBR-II) was built and operated by the DOE, and its predecessors, but did not undergo commercial licensing.

U.S. Sodium Fast Reactor Operational History

EBR-II (1963 – 1994)

FERMI 1 (1963 – 1972)

FFTF (1980 - 1993)
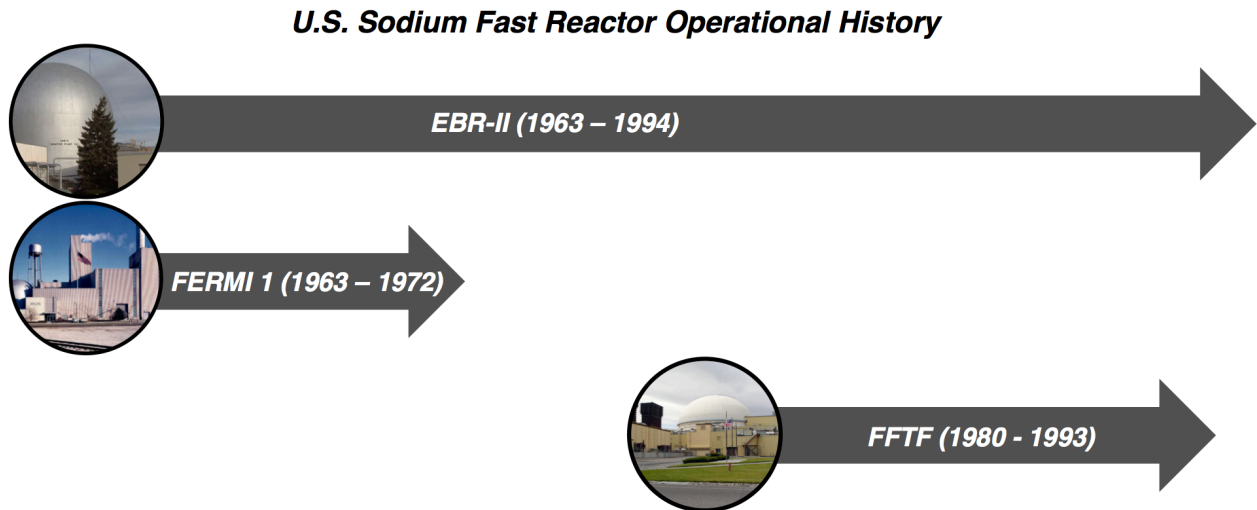
The Fast Flux Test Facility (FFTF) was also built and operated by the DOE, and its predecessors, and is the only SFR facility in the U.S. that received technical attention regarding a source term from the NRC, even though it was not commercially licensed[6]. The NRC received comments from the ACRS regarding proposed operation of FFTF in a letter dated November 8, 1978. Among the concerns of the ACRS was the satisfaction of the 10 CFR § 100 [17] guidelines regarding radiological consequences for a vessel melt-through accident. A semi-empirical analysis of aerosol transport was performed by the staff [18] to demonstrate that any credible core disruptive accident with containment venting would not violate the 10 CFR § 100 guidelines, where conservative assumptions regarding species, release fractions, and the timing of release were used to develop a containment source term. An empirically-developed Atomics International code, HAA-3, was then used to treat aerosol transport and behavior within and release from containment.

As shown in the following figure, throughout the 1970s, 80s, and 90s, several attempts were made at developing larger prototype SFRs as a step forward on the progression toward commercial deployment. Early source term analyses naturally evolved during preliminary licensing efforts and discussions between vendors and the regulatory body. The remainder of this section discusses source term development efforts for these proposed designs.

---

[6] Commercial SFR development was to be achieved through a gradual size (*i.e.*, power) increase in each design iteration, so that a facility on the scale of a commercial LWR could eventually be licensed and constructed. Because FFTF was one of these intermediate iterations, the ACRS was invited to provide an independent review of the facility in preparation for the expected licensing efforts for future designs, like the Clinch River Breeder Reactor.

U.S. Sodium Fast Reactor Design Development
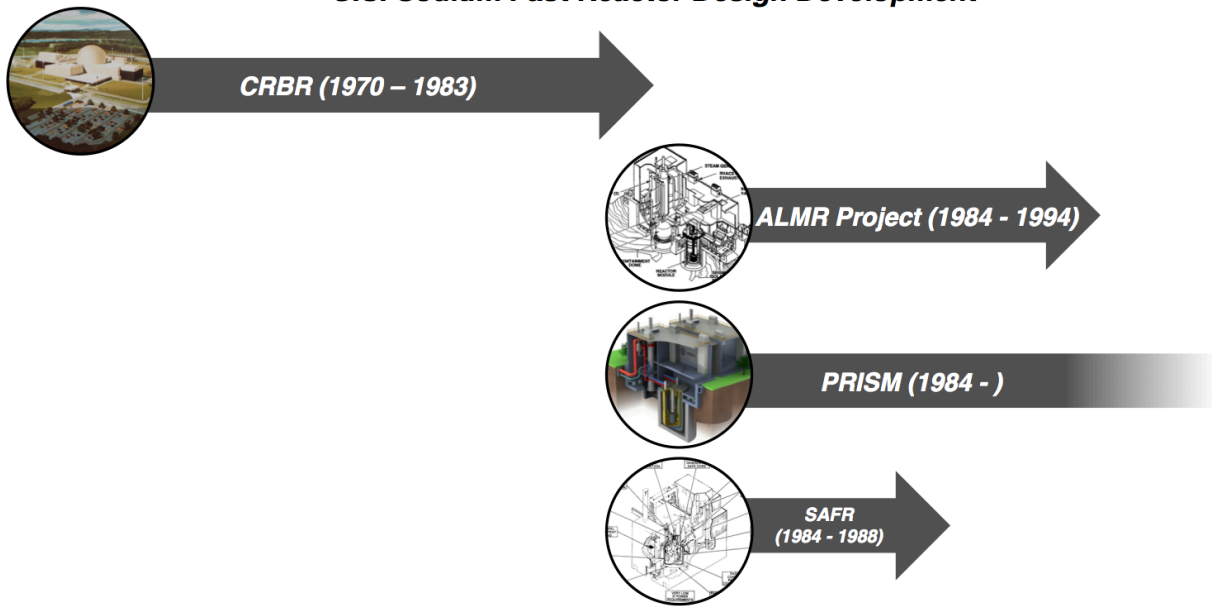
In the early 1970s, the AEC and nuclear industry began to jointly develop the Clinch River Breeder Reactor (CRBR), a full-scale prototype breeding SFR. Section 2.3.1 discusses the development of the CRBR design, the site suitability source term, and regulatory hurdles encountered. Following the cancellation of CRBR, the Sodium Advanced Fast Reactor (SAFR) and the Power Reactor Innovative Small Module (PRISM) concepts were proposed as part of the Advanced Liquid Metal Reactor (ALMR) project. Both designs completed significant design and analysis milestones, including early source term assessments. Section 2.3.2 discusses both projects' interactions with the NRC.

2.3.1 Clinch River Breeder Reactor

CRBR Summary
The CRBR project pursued the NRC licensing process, thereby completing a PSAR submittal, and receiving NRC comments. The siting source term was largely based on TID-14844, with a large instantaneous release of radionuclides into containment, and no credit for retention phenomena. The CRBR licensing process faced several challenges due to a regulatory path that included a focus on energetic accidents (HCDAs).

A joint endeavor by the research and development sector of the AEC[7] and U.S. nuclear industry, the CRBR project commenced in 1970. The primary objective of the CRBR project was to design, construct, and operate a prototype liquid metal fast breeder reactor (LMFBR) to demonstrate commercial viability of the technology. The CRBR was intended to be a loop-type,

---

[7] Prior to the Congressional dissolution of the AEC as part of the Energy Reorganization Act of 1974, the AEC was responsible for development and promotion of nuclear energy and nuclear weapons, as well as for regulation of the technologies. In 1974, during CRBR development, the AEC was split into the NRC, which is responsible for the regulation of nuclear energy technology, and the Energy Research and Development Administration, which would later become the DOE.

oxide-fuel SFR with breeding capabilities. The project was cancelled in 1983 due to escalating costs and licensing concerns.

During the course of the CRBR project significant effort was expended on safety analyses and preliminary fulfillment of applicable licensing requirements. As part of the CRBR project, a Preliminary Safety Analysis Report (PSAR) [19] was completed and submitted to the NRC consistent with guidance contained in the Standard Review Plan (SRP). Accordingly, the NRC issued NUREG-0968 [20], the Safety Evaluation Report (SER) for CRBR. The NRC approved the construction permit [20] and work began shortly before the project's cancellation.

As part of the CRBR PSAR, an analysis of radiological consequences affecting site suitability was performed. As per guidance from the NRC, a source term similar to the postulated release of TID-14844 was utilized; the resulting release fractions are shown in Table 2-6. As with TID-14844, it was assumed that significant radionuclide inventory was instantaneously released into the containment, and that containment leaked at the design basis rate. While CRBR was designed to utilize oxide fuel, it was also intended to operate as a breeder utilizing an inhomogeneous core (*i.e.,* blanket regions containing fissile or fertile material). Even though TID-14844 is based on oxide fuel phenomena, breeding presents a significant deviation from LWR oxide fuel behavior, as the fuel failure mechanisms and subsequent inventories may be different. The CRBR PSAR does note that the source term is overly conservative as it envelopes "conservatively hypothesized core-related events," but no alternative source term was proposed.

Table 2-6: Release Assumptions from CRBR PSAR [19]

Parameter	Assumptions
Release to containment	Noble gases: 100% Halogens: 50% (25% airborne) Solid fission products: 1% Plutonium: 1%
Containment leakage rate	Maximum allowable: 0.1% per day
Engineered safety features	No credit
Release period	Instantaneous

Much of the CRBR licensing process was centered on concerns regarding the risks of hypothetical core disruptive accidents (HCDAs), which are energetic events that could result from fuel melting. For an SFR, it is theoretically possible for the fuel to melt into a configuration that could cause a power excursion. While this was considered a bounding scenario for CRBR, HCDAs can be difficult to analyze as they could lead to fuel vaporization and the release of large amounts of energy that result in reactor vessel head failure and large sodium fires. Therefore the possible radionuclide release from such a scenario could be significant.

An HCDA was not considered to be part of the CRBR design basis, but design basis events potentially leading to an HCDA were of particular interest during regulatory review. At the time, many design features intended to prevent or mitigate HCDAs (core catchers, molten fuel dispersion implements, robust primary vessels, etc.) were considered in design revisions. Rigorous regulatory exploration of these events led to a lengthy and uncertain licensing process that would ultimately be one of the largest contributors to the project's cancellation. Following the CRBR licensing process, subsequent vendor designs (PRISM, SAFR) sought to remove HCDAs from those incidents considered credible for licensing through design changes.

2.3.2 The ALMR Program: PRISM and SAFR

ALMR: PRISM and SAFR Summary
Following the cancellation of the CRBR project, the SAFR and PRISM designs sought to use inherent and passive engineered features (such as metal fuel and pool-type primary systems) to improve safety and reduce the probability of severe accidents. Both design vendors submitted PSID documents, and received NRC responses. While PRISM utilized a large siting source term, similar to CRBR and TID-14844, SAFR pursued a source term intended to be representative of the melting of a single assembly. Neither source term analysis was mechanistic in nature, and the NRC commented on the simplifying assumptions made for both assessments.

Following the cancellation of the CRBR project in the early 1980s, the DOE initiated the ALMR program, leading to the rapid development of several commercial SFR designs by various vendors. These designs were a major departure from the CRBR precedent in that they sought to leverage inherent and passive safety mechanisms through a shift toward pool-type designs utilizing metal-alloy fuels, as opposed to the loop-type, oxide-fuel CRBR design of the 1970s.

These changes in design parameters were largely the result of an intentional emphasis on the passive safety features available to SFRs, which could be used to overcome the technical safety issues experienced during the licensing of CRBR. Submersion of the core in a large pool of coolant with no penetrations and excellent heat transfer properties creates a system with an extremely small probability of loss of coolant and the capability for cooling via natural convection. Maintaining the pool at ambient pressure also presents important safety implications by precluding energetic depressurization events from the primary system to containment. Additionally, a purposeful shift was made to metal-alloy fuel, which possesses several attractive benefits, as will be discussed in Section 3.1.1. These features were utilized as part of an endeavor to limit fuel melt propagation during accidents. In turn, the probability of HCDAs, and the challenging analyses and possible large radionuclide releases associated with them, could be removed from the realm of credible accidents based on phenomenological and probabilistic considerations [21]. Similarly, the focus on limiting melt propagation also helped reduce the probability of other severe accidents, such as core melt and slumping into an uncoolable geometry, with subsequent vessel melt-through.

During the ALMR project, the SAFR design by Rockwell International and the PRISM design by General Electric (GE) proceeded the furthest through the licensing process, including submission of Preliminary Safety Information Documents (PSIDs) to the NRC that included source term analyses. Accordingly, both designs received respective comments on the PSIDs from the NRC in Preliminary Safety Evaluation Reports (PSERs). In the absence of regulatory guidance for advanced reactor siting analyses, the SAFR and PRISM designs pursued significantly different source term strategies.

SAFR

Technical analyses of SAFR found that significant fuel melting was phenomenologically incredible due to the enhanced heat transfer capabilities of the metal-alloy fuel and the efficient decay heat removal systems. Therefore, the melting of a single assembly was proposed as a

bounding accident scenario for source term analyses. The analysis assumed the complete radionuclide inventory of the assembly was released to the primary sodium, but did credit retention in the sodium. The release fractions into the cover gas are shown in Table 2-7, where two cases of leakage into the containment were analyzed (100% leakage to containment over a day, and 100% instantaneous leakage into containment).

In the SAFR PSER [22], the NRC staff indicated that more information was required to support this type of deterministic source term, particularly with regard to the postulated holdup factors and fission product release fractions. The NRC staff also indicated that consideration should be given to power excursions initiated by sodium voiding, as they may lead to substantial fuel melting in the vessel.

PRISM

The site suitability analysis for PRISM utilized a different source term strategy from that of SAFR. Early in licensing efforts, GE indicated interest in development of an MST appropriate for metal-fuel pool-type SFRs. The NRC staff was supportive of an MST, but indicated that sufficient fuel qualification data should be available and fuel performance should be sufficiently well understood for normal and off-normal conditions. Also, the NRC stated that the transport of radionuclides through the appropriate barriers to the environs should be adequately modeled, calculations should be as realistic as possible so that the limitations of any barrier or mechanism are clear, and that bounding events should be selected that envelope the accident space and design-dependent uncertainties.

Due in part to the absence of metal-fuel data and the lack of credible fuel damage scenarios, in the PRISM PSID, conservative bounding assumptions regarding releases were utilized until additional MST research and development could be conducted. Consequently, the initial PRISM source term assumed complete core damage (including spent fuel in storage area), using a scaled oxide fuel radionuclide inventory, with little to no credit for radionuclide retention in the sodium pool and near instantaneous release to containment (essentially an HCDA scenario). In its review of the PRISM PSID, the NRC staff highlighted the use of an oxide fuel inventory for metal-fuel analyses as unrealistic, and stated that release fractions and retention/holdup factors would be reviewed in detail in future design reviews [23].

Table 2-7: Source Term Assumptions for the SAFR and PRISM Designs [22] [24]

Parameter	Assumptions	
	SAFR	*PRISM*
Accident Scenario	*Melting of single assembly*	*Whole core failure (including spent fuel in storage area)*
Release Period	*Instantaneous*	*Instantaneous*
Release Fractions	*Noble gases: 100% (of 1 assembly)* *Halogens: 0.01%* *Volatiles: 0.01%* *Plutonium: 7.8×10^{-7}%*	*Noble gases: 100% (of whole core)* *Halogens: 0.1%* *Volatiles: 0.1%* *Transuranics (Pu): 0.01%*
Release into	*Cover gas*	*Containment*
Engineered Safety Features	*No credit*	*No credit*
Containment Leakage Rate	*Design basis*	*Design basis*

2.4 Current SFR Vendor Designs and Reference Design

This section provides an overview of current SFR vendor designs, and establishes a generic metal-fuel pool-type SFR configuration, which is used as the reference design throughout the remainder of this document. The metal-fuel pool-type SFR configuration was chosen as it is representative of the designs proposed by the four SFR vendors currently pursuing deployment in the U.S. (General Electric-Hitachi, TerraPower, Advanced Reactor Concepts (ARC), and Toshiba). The specific vendor designs are reviewed in detail at the end of this section.

In a pool-type SFR configuration, components typically exterior to an LWR reactor pressure vessel, such as primary coolant pumps and intermediate heat exchangers, are contained within the sodium pool, as illustrated in Figure 2–2 [25]. An inert cover gas is located above the pool region, since sodium is reactive with oxygen and water. The entire vessel is sealed by an upper head to prevent leakage of the cover gas and ingress of air, and the cover gas region is typically maintained at near atmospheric pressures (hence, the primary sodium is also at near atmospheric pressure). Typical core inlet/outlet temperatures are approximately 350°C/500°C, respectively.
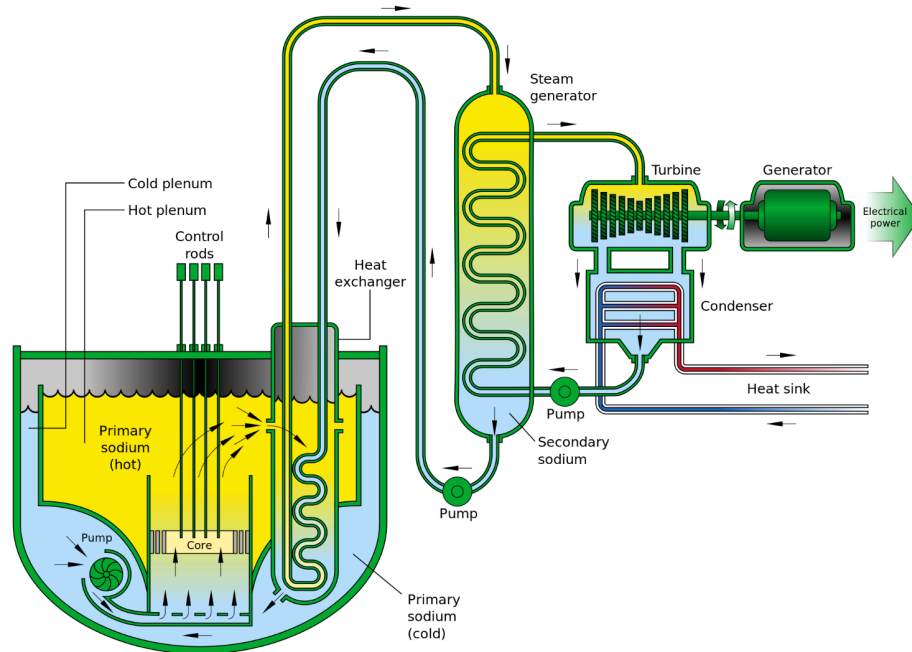
Figure 2–2: Schematic of Pool-type SFR Plant [25]

Pool-type SFRs present inherent safety benefits due to the encapsulation of the entirety of the primary coolant within a reactor vessel and guard vessel lacking any penetrations below the primary sodium level. The inclusion of the guard vessel precludes some conventional LWR accident scenarios, such as LOCAs, and removes many containment bypass mechanisms. For the reference design here, the containment configuration in Figure 2–3 will be used. The containment boundary consists of two main elements:

1) **Guard Vessel** – A secondary vessel that surrounds the reactor vessel and can capture the primary sodium in an inert atmosphere in the low probability event of a

reactor vessel leak. The guard vessel is designed to ensure the reactor core will remain submerged in sodium, even if a breach in the reactor vessel occurs.

2) **Upper Containment** – For the reference design, the upper containment resembles a smaller version of the large dry containment seen in PWRs. A steel liner is typically placed on the inner side of the upper containment, with a concrete structure around it. The pressure rating of the upper containment is usually lower than LWR containments since SFRs are low-pressure systems. However, external threats may dictate upper containment specifications.

The reactor vessel and guard vessel are usually supported at the top via a circumferential flange, with no supportive structure below the vessels. An inert gas is usually placed between the two vessels to prevent a sodium fire in the event of a reactor vessel leak.
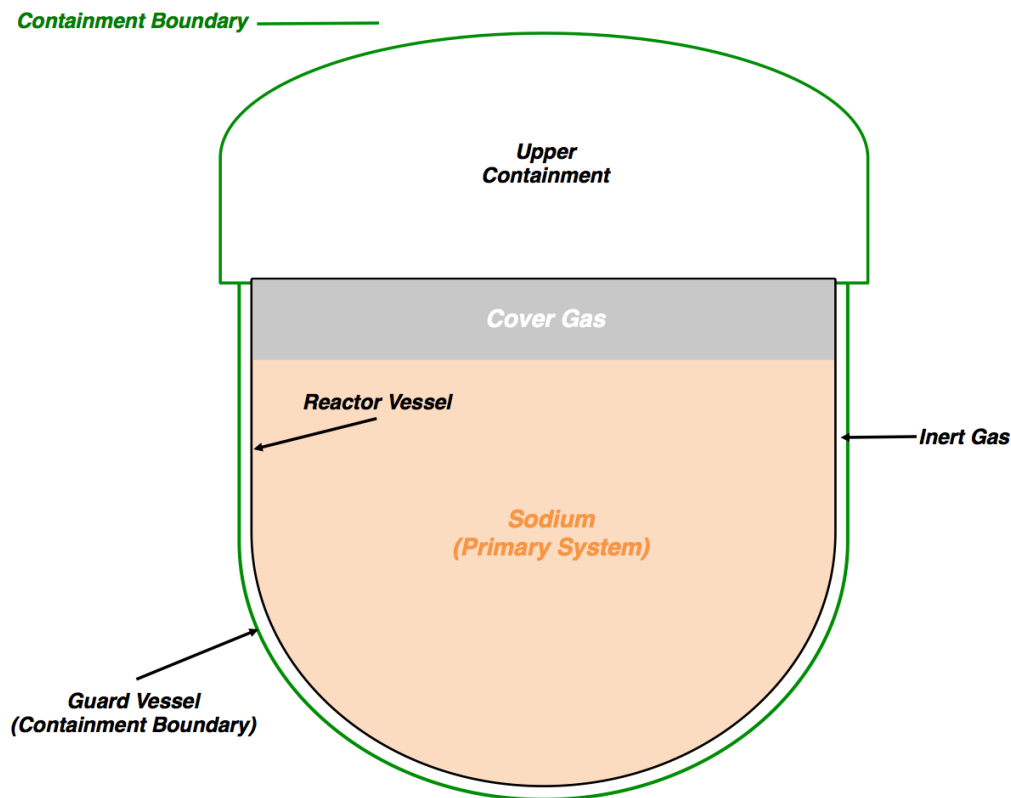
Figure 2–3: Simplified Pool-Type SFR Containment Boundary Variation

The containment configuration used as the reference design here is one containment option, as there have been multiple containment configurations proposed for past designs. For example, early PRISM design proposals did not include a conventional upper containment, while later variations included a small steel dome above the reactor vessel referred to as the head access area [24]. Also, the containment design depicted in Figure 2–3 is not unique to metal fuel, as oxide fuel pool-type SFRs may use a similar layout.

An example reactor building, including the reactor vessel and upper containment are shown in Figure 2–4 [26]; this is one possible configuration, but is representative of most conventional metal-fuel pool-type SFR designs. The annular region above the reactor contains piping rooms for

the intermediate circuit and auxiliary systems, such as the primary sodium cleanup system. The reactor building is placed on seismic isolators, with chimneys for passive decay heat removal systems on opposite sides.
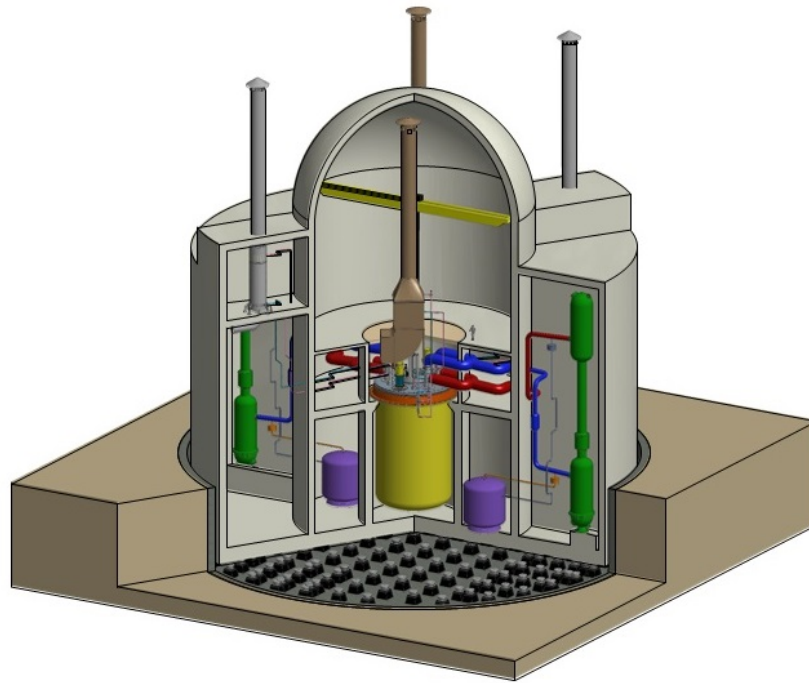
Figure 2–4: Example Pool-Type SFR Reactor Building [26]

Although the entirety of the primary sodium coolant is contained within the reactor vessel, there are several reactor vessel head penetrations related to auxiliary and support systems (described in detail in Section 3.2). For example, a primary sodium purification system transports a small quantity of sodium through the reactor vessel head to a cold trap that removes impurities. To prevent the inadvertent release of radionuclides during normal operation and to maintain the purity of the cover gas, a cover gas cleanup system is typically utilized, where the majority of the associated piping and retention/cleanup tank are usually located outside of containment with penetrations to the reactor cover gas region. The sodium in the intermediate loop can also become slightly activated as it travels through the activated primary sodium pool; however, it should be emphasized that the radionuclide inventory is far below that of the primary circuit sodium. There are also sealed penetrations in the vessel head that may present potential leakage pathways from the cover gas region, where these penetrations are utilized for instrumentation, control rod drives, refueling ports, and associated piping.

The reference design for this work is assumed to have metal-alloy fuel, which is common in all current U.S. SFR designs. Metal fuels, described in Section 3.1.1, present certain inherent safety features over oxide fuel [27]. Metal alloys have attractive material properties for safety applications such as beneficial inherent feedback effects and favorable behavior during transient overpower events. Unlike oxide fuels, which can react with the sodium and result in fuel dissolution into the coolant, metal fuels are nonreactive with sodium. This limits the propagation of pin failure and release of radionuclides. It should be noted however that mixed oxide fuels and loop-type SFRs have been and are still being utilized internationally.

To illustrate the variations between the four current SFR vendor designs, key parameters are shown in Table 2-8. All of these reactors utilize a pool-type configuration with a metal-alloy fuel, but are sized differently. It should also be noted that the Traveling Wave Reactor (TWR) design plans to utilize vented fuel, whereas the other designs under consideration use a traditional sealed pin design (which is discussed in Section 3.1.1). Several of the designs are pursuing extended refueling cycles.

Table 2-8: Key Characteristics of Vendor Proposed SFR Designs[1]

Characteristic	4S (Toshiba)	TWR (TerraPower)	PRISM (GE-Hitachi)	ARC-100 (ARC)
Power	$30MW_t$ /$10MW_e$ or $135MW_t$ /$50MW_e$	$1200MW_t$ /$500MW_e$	$425MW_t$ /$138MW_e$ per module	$260MW_t$ /$100MW_e$
Refueling cycle	30yr/10yr	50yr	26mo	20yr
Vented fuel	No	Yes	No	No
Vessel size	3.6 m dia	13.3 m dia 17.65 m h	5.74 m dia 16.94 m h	7 m dia 15.63 m h

[1]All designs utilize a pool-type reactor configuration with metal-alloy fuel.

For comparison, key characteristics of past SFRs operated in the U.S. are shown in Table 2-9. These designs explored a range of SFR configuration options, as varying primary system configurations and fuel types were utilized. Of the SFRs operated in the U.S., EBR-II shares the most similarities with the current vendor designs.

Table 2-9: Key Characteristics of SFRs Operated in the U.S.

Characteristic	EBR-II	Fermi 1	FFTF
Power	$62.5MW_t$ /$20MW_e$	$200MW_t$ /$61MW_e$	$400MW_t$
Fuel Type	Metal	Metal (1963-66) Oxide (1970-72)	Mixed oxide
Primary Configuration	Pool	Loop	Loop
Vented Fuel	No	No	No
Criticality/Shutdown	1963/1994[1]	1963/1972	1980/1993

[1]EBR-II achieved dry criticality (no sodium in the primary system) in 1961, and wet criticality (sodium in the primary system) in 1963.

3 SFR Sources of Radionuclides

This section reviews the major sources of radionuclides for a metal fuel, pool-type SFR (utilizing the base design described in Section 2.4). First, the fuel is examined, including spent fuel handling and storage. This is followed by a review of radionuclides in the primary sodium and cover gas, along with their associated cleanup systems.

3.1 Fuel

3.1.1 Types of Fuel

> **Types of Fuel Summary**
> *Metal fuel has many desirable safety characteristics such as its compatibility with the sodium coolant, excellent thermal conductivity, and low operating temperature. Although there is a potential for fuel-cladding interaction at elevated temperatures, inherent reactivity feedbacks provide passive reactivity reduction with increasing temperatures. Metal fuel experience in the U.S. is substantial, including irradiation experience and accident condition testing. The exact composition of metal fuel can vary, as SFRs can be designed to be burners or breeders*

In an attempt to match the high burnup levels of commercial LWR fuel, multiple iterations of metal fuel design have evolved throughout the history of the various SFR-related programs. Initial metal fuel designs in the 1960s used unalloyed uranium or plutonium driver fuels with relatively high smear densities (the areal density of as-fabricated fuel within as-fabricated cladding). Early discovery of challenges associated with the fabrication and utilization of unalloyed fuel, however, led to the testing of various alloys in metal fuel. Inclusion of alloys in a metal fuel tends to increase its mechanical strength, improve its chemical stability under irradiation, and aid in the fabrication process and performance during off-normal operation via the modification (typically an increase) of liquidus and solidus temperatures.

The first EBR-II driver fuel utilized a U-5Fs alloy; Fs, or fissium is a composition[8] of simulated noble metal fission products produced in equilibrium from the EBR-II fuel recycling process. By 1970, zirconium (Zr) gained momentum as the preferential metal fuel alloy and was soon suggested for use in EBR-II. The use of zirconium would significantly reduce the interdiffusion of fuel and clad components, where this fuel-clad chemical interaction was a significant contributor to fuel life reduction resulting from clad thinning [28].

Although the second and third iterations of EBR-II driver fuel still utilized the fissium alloy, in the mid-1980s, the ALMR program selected U-Pu-Zr as reference fuel [27]. Plutonium and/or uranium alloyed with zirconium is typically used as the reference fuel in most modern SFRs. The operating experience with metal fuel is quite substantial, as shown in Table 3-1, with extensive irradiation experience (>100,000 fuel rods) and multiple qualification and safety tests. Comprehensive reviews of fast reactor fuel experience [29] [30] suggest that the existing metal fuel database is sufficient to make a safety case for use of metal fuel in a demonstration or

---

[8] 2.4 wt% Mo, 1.9 wt% Ru, 0.3 wt% Rh, 0.2 wt% Pd, 0.1 wt% Zr and 0.01 wt% Nb [29]

prototype facility, provided that the fuel composition and burnup are expected to be within the envelope of the available database.

Table 3-1: Summary of U.S. Metal Fuel Experience [30]

Characteristic	No./Type of rods	Clad	Burnup	Comments
Driver Fuel Operation	~90000 U-Fs >30000 U-Fs ~13000 U-Zr	304 LSS 316 SS 316 SS	1-3 at.% 8 at.% 10 at.%	
Through Qualification	U-Zr	316 SS, D9, HT9	10 at.%	EBR-II, FFTF
Demonstrated Burnup Capability and Experiments	U-Pu-Zr	D9, HT9	10-20 at.%	EBR-II, FFTF
Safety and Operability	U-Fs & U-Pu-Zr/U-Zr			RBCB tests, discussed in Section 4.3.3
	9 U-Fs 6 U-Zr/U-Pu-Zr	316 SS D9/HT9		TREAT tests, discussed in Section 4.3.4

In an effort to close the fuel cycle, fast reactors have traditionally been designed as either a burner or breeder reactor. Burner reactors, which transmute, or burn, transuranics, are intended to reduce LWR spent fuel and weapons stores and prevent some degree of proliferation by eliminating the need for conventional enrichment techniques. While this process is possible in the thermal spectrum, transmutation via fissioning is most effective in the fast spectrum. In a burner reactor, fuel is generally homogenously distributed, although enrichment zones may be utilized to flatten the power profile. Some designs have considered the utilization of uranium-free metal fuel in their burner SFR to reduce the existing plutonium stockpiles and high-level waste [31]. The proposed TRU-Zr fuel is ideal because it does not produce any additional TRU due to the absence of uranium.

An alternative to the burner configuration, a breeder reactor utilizes blankets of fissile/fertile material to effectively create more fissile material than it consumes. The concept of breeding was an attractive design alternative in the 1960s, when it was believed that uranium supplies might be inadequate to meet near-term needs. The fast spectrum is an ideal choice for breeding due to its ability to utilize either the uranium or thorium fuel cycles. Breeding still occurs without a blanket configuration, but at a reduced efficiency. Today, various SFR designs with varying conversion rates are being pursued (included the "breed and burn" concepts like TWR), but most aim for a near break-even core design with a conversion rate of approximately one.

Metal fuels are chemically compatible with the sodium coolant, as they do not react with the sodium, which is a possibility with oxide fuel. In addition, metal fuels have small internal temperature gradients, as metals have excellent thermal conductivity. Beyond the chemical and mechanical benefits, metal fuel is also neutronically favorable, due to its inherently negative reactivity feedback. Axial expansion of the fuel and radial expansion of the core introduce negative reactivity with increasing core temperatures. Also, the change in Doppler broadening with reduced temperature is more favorable than with oxide fuel, due to a significantly lower fuel

operating temperature. The ability of these feedback effects to successfully shut down the core during unprotected (without scram) transients has been demonstrated in EBR-II tests [32].

Metal fuel pin designs are similar to commercial LWR fuel pin designs, in that a fuel slug is entirely contained within sealed cladding, and there is some void, or plenum, within the pin that allows for fission gas release and slug expansion. Unlike oxide fuels, which are usually formed into pellets, metal fuels are injection cast as one continuous piece. A metal fuel pin is also fabricated with sodium around the fuel slug, so the fuel slug remains thermally bonded to the clad prior to operation; a schematic of this configuration is shown in Figure 3–1. Although bond sodium helps keep the fuel temperature low during the early stages of operation, upon sufficient irradiation and heating, the fuel slug will expand such that it is in contact with the cladding.
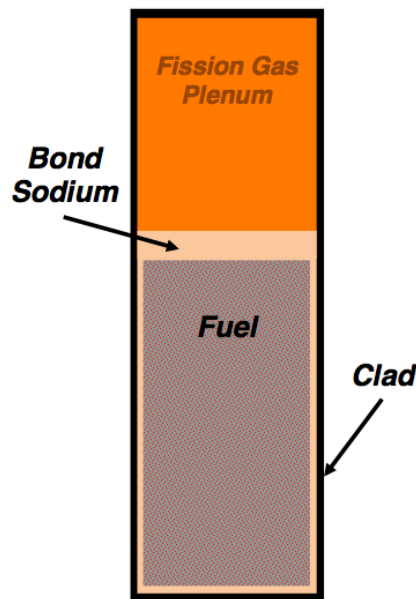

Figure 3–1: Simplified Schematic of Typical Metal Fuel Pin

For metal fuel, the fission gas plenum is a substantial portion of the pin to prevent cladding breach resulting from overpressurization. This design feature evolved from lessons learned regarding slug expansion, fission gas buildup, and clad failure that occurred early in the operation of EBR-II. While the early inclusion of alloys in metal fuel proved advantageous, it did not achieve the high burnup goal of nearly 20 at% set by the fast reactor program to make the new reactors competitive with the existing LWR technology. Cladding breach occurred at relatively low burnups, largely due to creep rupture of the cladding induced by fuel swelling and pin pressurization from fission gas buildup [33] [34]. In early EBR-II operations, it was theorized that the current fuel design swelled approximately 30% before fuel-clad contact occurred and in-pin fission gas movement would be prevented [35]. Therefore, the third iteration of EBR-II driver fuel utilized a reduced smear density of 75%, which allowed fission gases produced during burnup to move within the pin (through interconnected pores in the fuel matrix) and transfer to the fission gas plenum. As a result, the fuel-cladding mechanical interaction was significantly reduced, preventing premature cladding breach.

There are modern, domestic SFR designs proposing the use of a vented fuel to prevent any overpressuration- or swelling-induced clad breach. To achieve longer refueling cycles, fuel must be able successfully withstand higher burnups and remain intact. As an alternative to increasing the size of the fission gas plenum, which may be prohibitively long to achieve very high burnups, some vendors are proposing the use of vents in the pin that will continuously release fission gas into the sodium coolant. However, in this case, a robust on-line primary purification system would be required to assure that the activity of the coolant is within specified limits. Such a fuel choice does impact the potential release of radionuclide and the MST development, as there is a reduction of fission gas inventory in the fuel pin, but an increase of radionuclides in the primary sodium and stored in the cleanup system. It should be noted that the existing SFR operation database may be unable to qualify vented fuels, depending on the design of the fuel and the extent of deviation from past fuel designs and operational experience.

Of the phenomena affecting metal fuel lifetime, fuel slug swelling and fission gas release are relatively well understood such that fuel is now well protected against these failure mechanisms via design features. For metal fuel, the most dominant failure mechanism at elevated temperature tends to be produced by the fuel-clad chemical interactions that shorten the life of the cladding. The interdiffusion of elements from the fuel (uranium, plutonium, and rare earth fission products) and cladding (iron) results in the formation of a eutectic mixture with a lower melting point than any of the individual constituents. The actual timing of pin rupture is highly dependent on the integral thermal stress the material experiences; at 650°C, clad may fail on the order of hundreds of hours, while at elevated temperatures near the boiling point of sodium (883°C), failure due to eutectic penetration of the cladding could occur on the order of minutes [36].

3.1.2 Core Design

> **Core Design Summary**
> Metal fuel SFR core designs generally use very tightly packed hexagonal fuel assemblies, with cans that prevent coolant flow between assemblies. Wire wraps on the pins ensure that the pins do not contact and experience wear due to vibration, while allowing compact assembly configurations. Spent fuel can also be stored in the primary sodium in an area around the core.

A typical SFR core is composed of a series of subassemblies[9], each containing multiple pins. For SFRs capable of producing higher power, the number of subassemblies within a core can be on the order of hundreds, with each subassembly containing up to several hundred pins. For example, a 30 MW$_{th}$ SFR may contain approximately 20 subassemblies, each with 170 fuel pins, whereas a 1000 MW$_{th}$ SFR may contain approximately 230 subassemblies, each with 270 fuel pins. Except at the top and bottom, each subassembly is closed, thereby isolating flow through the subassembly and preventing cross flow between subassemblies. To maintain separation between neighboring pins while maximizing the area available within a subassembly, fuel pins are typically wrapped in wire. An example of this is shown in Figure 3–2, which shows a wire wrapped fuel pin from the Advanced Burner Test Reactor (ABTR) conceptual design [37].

---

[9] A *subassembly* in an SFR and an *assembly* in an LWR are analogous terms describing rod bundles and are interchangeable. Historically, the term subassembly has been used in the SFR community, but use of the LWR terminology assembly has become more commonplace.
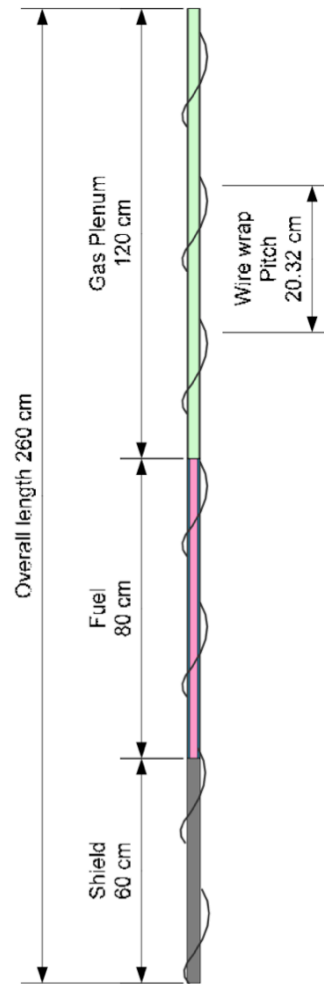
Figure 3–2: Schematic of Wire-wrapped Fuel Pin from the ABTR Conceptual Design [37][10]

Very early SFR designs, such as Fermi 1, utilized square fuel arrays in the core layout, as seen in Figure 3–3. During the initial years of the fast reactor program, hexagonal subassemblies, as shown in Figure 3–4, were pursued in parallel with the square lattices; EBR-II and FFTF were designed to utilize the hexagonal arrays. Ultimately, hexagonal subassemblies are the foundation of the primary core layout configuration utilized today, due largely to their favorable neutronic characteristics and the development of the wire wrapped pin, which enabled tighter lattices. Fast reactor fission efficiency increases with the hardening of the neutron spectrum, which can be accomplished by increasing the fuel volume fraction relative to the coolant and structure density. Mechanically, this requires tight fuel lattices with a large pin to coolant/structure area ratio. A tightly packed hexagonal array maximizes the fuel density, while application of wire wraps around the pins prevents vibration-induced clad damage and local hot spots. An example of a full core layout using hexagonal subassemblies is shown in Figure 3–5.

---

[10] The inclusion of shielding is common above and/or below the fuel slug, depending on the vendor design.
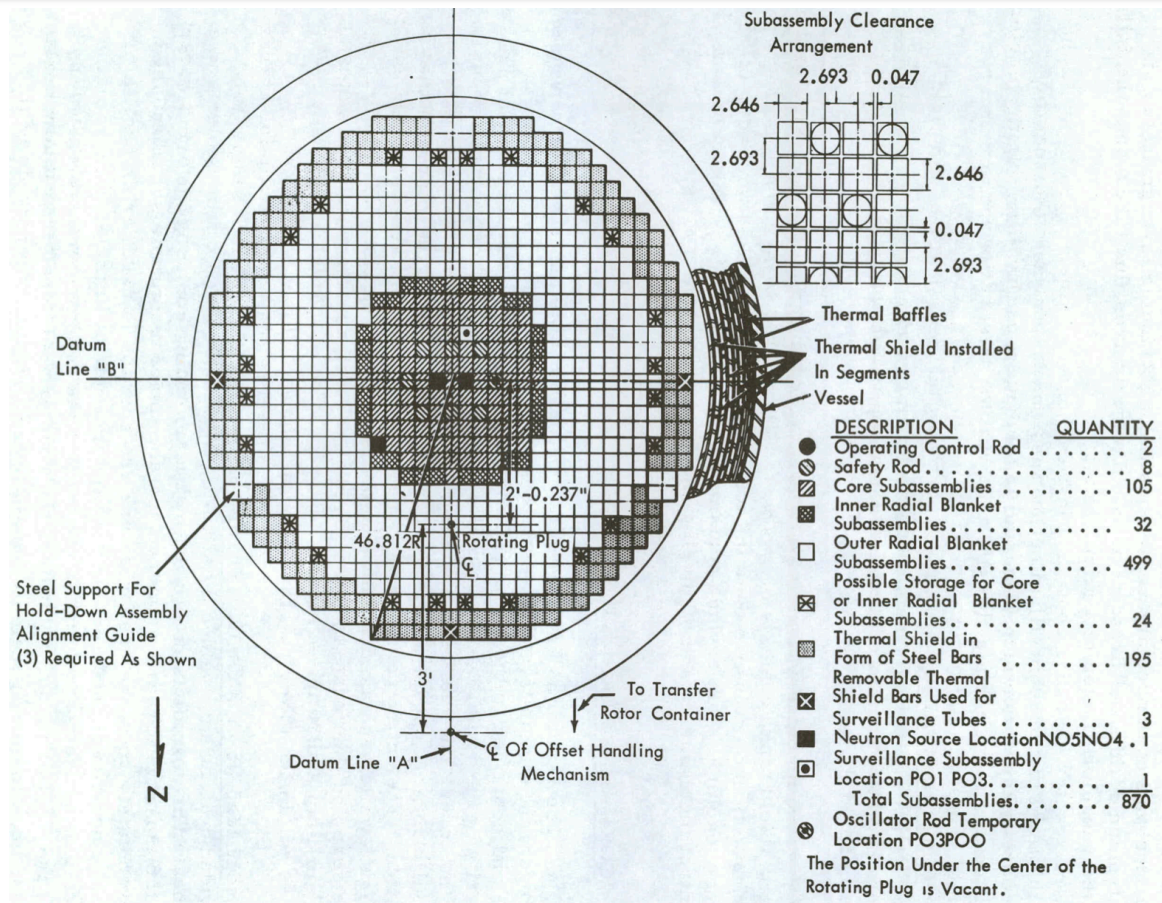
Figure 3–3: Schematic of Fermi 1 Core Layout using Square Subassemblies [38]
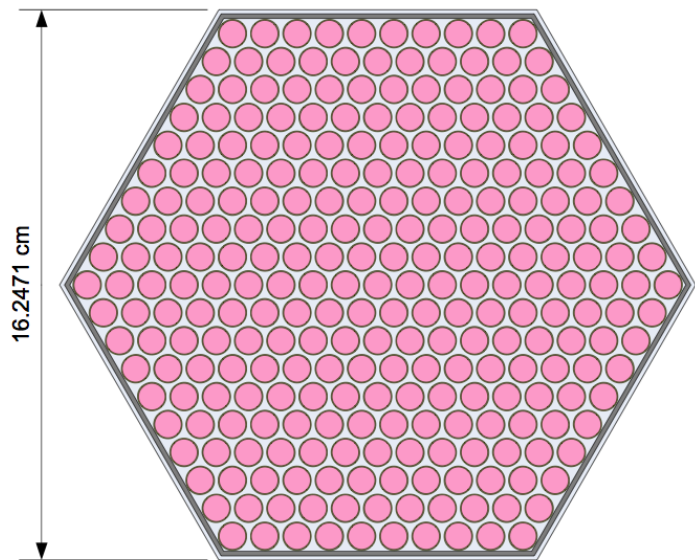

Figure 3–4: Example of Hexagonal Assembly from the ABR-1000 Conceptual Design [39]
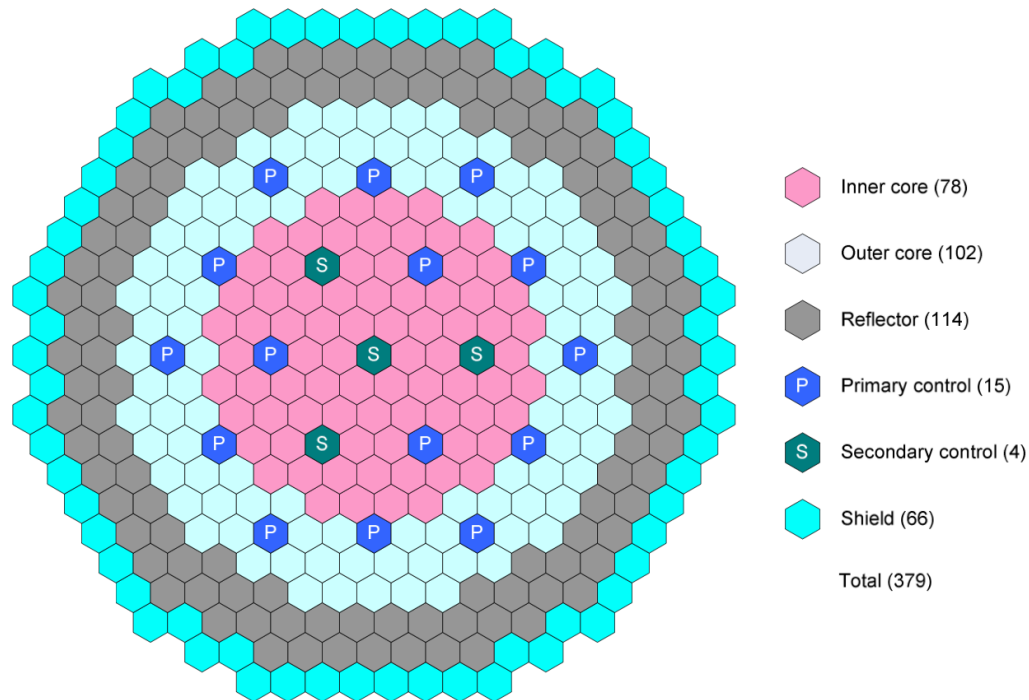
Figure 3–5: Core Layout Diagram using Hexagonal Assemblies from ABR-1000 Conceptual Design [39]

After depletion, some proposed SFR designs intend to first store spent fuel at the periphery of the core, rather than in a spent fuel pool or onsite cask with active cooling. Stored beyond the reflector where it will not significantly influence the neutronic characteristics of the core, in-pool storage allows for the use of bypass flow from the core or natural circulation in the primary sodium pool to remove decay heat. This allows a decrease in radioactivity and decay heat before the spent fuel is removed from the reactor vessel.

3.1.3 Spent Fuel Handling

> **Spent Fuel Handling Summary**
> There are several ways spent fuel can be transported from the vessel of an SFR. In the U.S., both gas-cooled moving casks and sodium pot moving casks have been used to transfer fuel from the reactor vessel to a storage area.

The possibility of radionuclide release resulting from the handling and storage of spent fuel may also affect the development of a plant MST. There have been various procedures used for removing and transporting spent fuel from a sodium reactor. Ref [40] provides a comprehensive review of past designs; a summary of the most common procedures is provided here as background for the source term discussion in Section 5.2.

First, as discussed in the preceding section, many metal-fuel SFR designs incorporate spent fuel storage in the primary sodium pool. Here, spent fuel is stored for some period of time until the decay heat or radioactivity has reached an acceptable level. The spent fuel handling process becomes more complex when the assemblies are removed from the sodium pool and transported to a longer-term storage area. In general, the process can be broken down into several steps:

1) Spent fuel is moved from the spent fuel storage area around the core to a basket or transfer device.
2) The basket or transfer device is raised from the primary sodium pool, through a head access port or shoot, and into a cask or transport vessel.
3) The cask or transport vessel moves the spent fuel to a longer-term storage area.
4) Spent fuel is moved from cask or transport vessel to longer-term storage.

The most interesting portion of this process, from a source term perspective, is after the spent fuel has left the primary sodium pool and is being transported in the cask or transport vessel since the fuel is now outside of the barrier boundaries of the primary system and containment. There are three popular methods for this task:

- **Gas-cooled moving casks** – A cask where the spent fuel is placed in an inert gas environment, which can use passive or forced gas cooling.

- **Sodium pot carrying casks** – A cask where the spent fuel is placed in sodium, which can be passively or actively cooled.

- **Sodium pot transfer shoots** – Similar to the sodium pot cask, however, the spent fuel transport device moves through an enclosed shoot connected to the reactor vessel, rather than leaving the reactor vessel and using a separate transport vehicle.

In the U.S., EBR-II utilized a gas-cooled moving cask, while FFTF and Fermi 1 used sodium pot carrying casks. The source term phenomena related to a spent fuel handling accident is dependent on the type of transport vehicle used, and is explored in Section 5.2.

3.2 Primary Sodium and Cover Gas Systems

This section provides an overview of the radionuclide inventory found in the primary sodium and cover gas region of a pool-type SFR during normal operation, along with a description of the systems used to remove radionuclides and impurities from each.

3.2.1 Primary Sodium Radionuclides

> **Primary Sodium Radionuclide Summary**
> U.S. experience with pool-type SFRs demonstrates that sodium activation (particularly ^{24}Na) dominates primary sodium radioactivity during operation. However, its short half-life means that it quickly decays after shutdown. With pin breaches aside, tritium and ^{22}Na become the next major sources, with radionuclides from corrosion products and sodium impurities likely falling at lower levels. Any pin breaches will likely result in the presence of iodine and cesium in the primary system due to their high solubility in sodium.

The primary sodium within the pool of an SFR may contain radionuclides from several sources, including sodium activation, tritium production, corrosion product activation, and fission products from failed fuel pins.
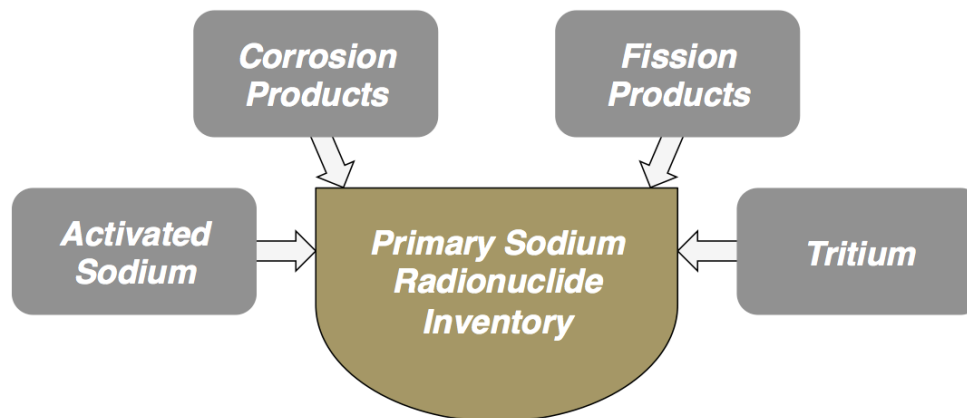
Primary Sodium Activation

Natural sodium is composed entirely of the stable isotope ^{23}Na. Within the pool of an SFR, ^{24}Na, which has a ~15 hour half-life, is created from the reaction ^{23}Na(n, γ) ^{24}Na. In smaller quantities, the longer-lived isotope ^{22}Na (half-life = 2.6 years) is produced by $(n,2n)$ reactions and from ^{23}Ne (half-life = 38 sec) produced from ^{23}Na(n, p) ^{24}Ne reactions [41]. Impurities in reactor-grade sodium are not considered to have a significant effect on the activity levels [41].

Tritium Production

Tritium is produced by ternary fission in the fuel and can be found in the primary sodium due to subsequent diffusion through the cladding [42]. While usually not a major operational concern, it is highly mobile, with the ability to diffuse through structure. The level of tritium within the primary circuit depends greatly on the operational conditions, plant layout (such as the ability for diffusion through a steam generator), and hydrogen sources. For most pool-type SFRs, tritium may escape the primary circuit through transport to the cover gas, through the heat exchangers to

the intermediate sodium loop, through the vessel or auxiliary piping walls, and by removal through the cold trap (as will be discussed in Section 3.2.2).

Corrosion Products

Since the majority of core components have historically been, and still are, constructed using type-300 stainless steel, the activation of corrosion products is likely. In particular, the creation of ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{58}Co, ^{60}Co, and ^{182}Ta are all possible, with a focus on ^{54}Mn, ^{58}Co, and ^{60}Co during shutdown since they have longer half-lives [41].

Fission Products and Fuel

Fission products and fuel may also be found in the primary sodium as a result of failed fuel pins. These are stochastic fuel pin failures that have occurred due to structural imperfections, not due to core conditions outside of normal operation (such as during accident sequences). Of greatest concern is the release of cesium and iodine, which are both highly soluble in the sodium (the retention of these elements within the fuel and solubility in sodium is discussed in greater detail in Section 4.2). Figure 3–6 reviews the maximum activity levels that were measured in the primary sodium at both EBR-II and FFTF[11] [41].

The activity of ^{24}Na is by far the largest contributor during operation, but quickly diminishes after shutdown due to its short half-life. The other main sources of radionuclides (^{22}Na, ^{131}I, ^{137}Cs, ^{134}Cs, and ^{3}H) are comparable to each other in their contribution. The higher levels of ^{131}I and cesium found in EBR-II were due to a greater number of pin breaches compared to FFTF [43]. The level of tritium found in FFTF is likely higher than what would be seen in a power plant since FFTF did not have a steam-water circuit, which reduced the level of hydrogen in the primary system and resulted in the cold trap being less efficient at tritium removal (as will be discussed in the following section). Activity from radionuclides associated with corrosion products and sodium impurities are below the minimum level shown in Figure 3–6 (< 1 MBq/kg) [43].

---

[11] Even though FFTF is a loop-type, oxide-fuel SFR, information on the radionuclide inventory found in the primary sodium is still helpful for illustrating the magnitude of radionuclides in an SFR system.

Isotope	^{24}Na	^{22}Na	^{131}I	^{137}Cs	^{134}Cs	^{3}H
Half-life	15 h	2.6 yr	8 d	30 yr	2.2 yr	12 yr
EBR-II	110000	5.2	11	13	1.6	X*
FFTF	370000	22	1.8	8.1	5.6	11
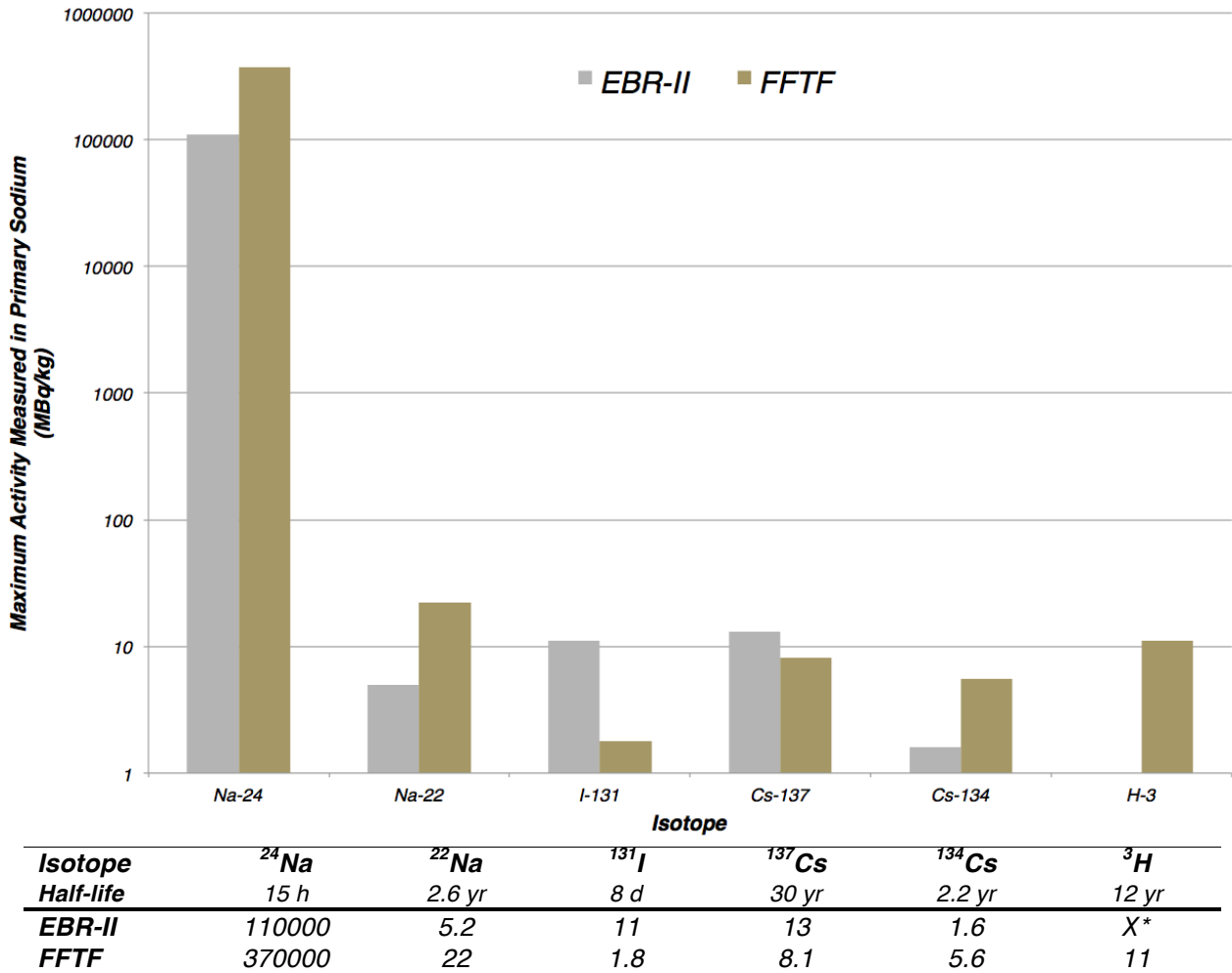
Figure 3–6: Maximum Activity Levels (MBq/kg) Measured in Primary Sodium [43]
* Not applicable due to the presence of Li containing experiments

3.2.2 Primary Sodium Purification and Cleanup Systems

Primary Sodium Purification Summary
SFRs use cold traps to remove impurities from the primary sodium. While cold traps are designed to remove oxygen and hydrogen to prevent component corrosion, they can also retain tritium (depending on the hydrogen concentration in the sodium) and some cesium. Dedicated cesium traps were also developed and used at EBR-II and FFTF. The cold traps may be placed within the primary pool or in a room within containment, which can have implications on the release of radionuclides were a break to occur in the system.

Primary sodium purification/cleanup systems are used to reduce oxygen and hydrogen impurities that would normally cause corrosion (and therefore contamination) of primary system components. While these systems are designed for this reason, they can also assist in the reduction of other radionuclides present within the primary circuit. The most common sodium cleanup method is through the use of cold traps, which cool the sodium below the saturation temperature of the impurities, allowing them to crystalize on the walls of the cold trap vessel or on a stainless steel mesh within the trap.

A schematic of the EBR-II primary sodium purification system is shown in Figure 3–7. Sodium is pumped from near the bottom of the reactor vessel to a surge tank, which prevents the siphoning of sodium should a break occur in the cleanup system. In this design, the cold trap is located inside of the containment boundary. The purified sodium is reheated by the incoming sodium before returning to the reactor vessel.
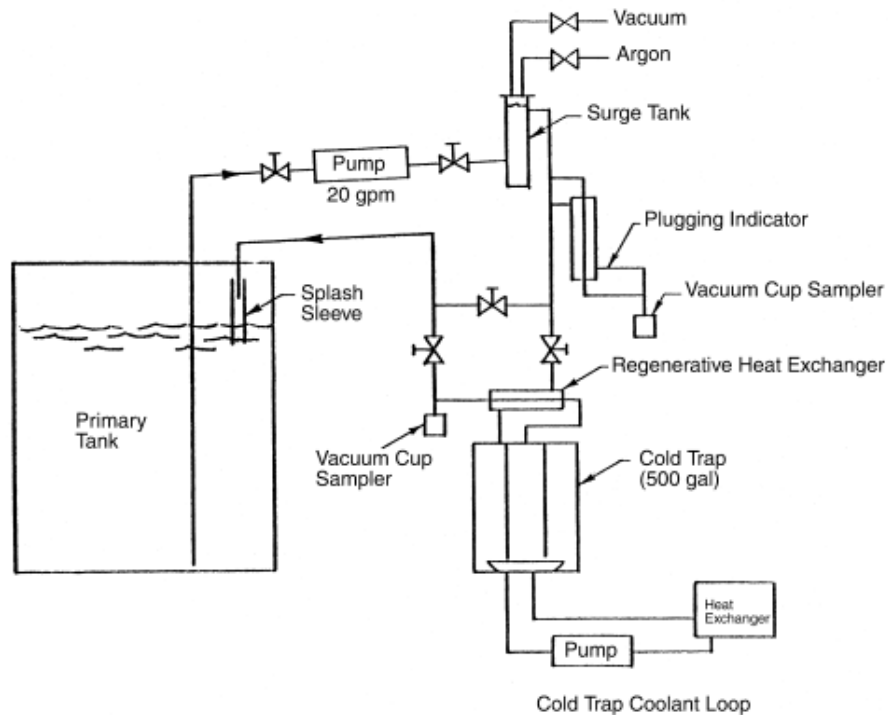
Figure 3–7: EBR-II Primary Sodium Purification System [44]

Other reactor designs have placed the cold trap within the reactor vessel and primary sodium pool. For example, Figure 3–8 illustrates such a design for the SAFR reactor, where the cold trap is within the primary sodium pool, and cooling systems reach the cold trap through a penetration in the reactor vessel head. The location change does have possible MST implications if a release from the cold trap were to occur. This is discussed in greater detail in Section 5.3.

Operational experience with the cold traps at EBR-II demonstrated an ability to capture the majority of oxygen and hydrogen within the primary sodium in a single pass through the cold traps. However, of more interest here, a large portion (between 45-85% per pass) of the tritium was also removed [42]. While the cold traps were unable to remove the tritium directly, since tritium does not reach the saturation level even in the cold trap, it can co-crystalize with other hydrogen present in the primary system or undergo an isotopic exchange with hydrogen (NaH) already deposited in the cold traps [41].
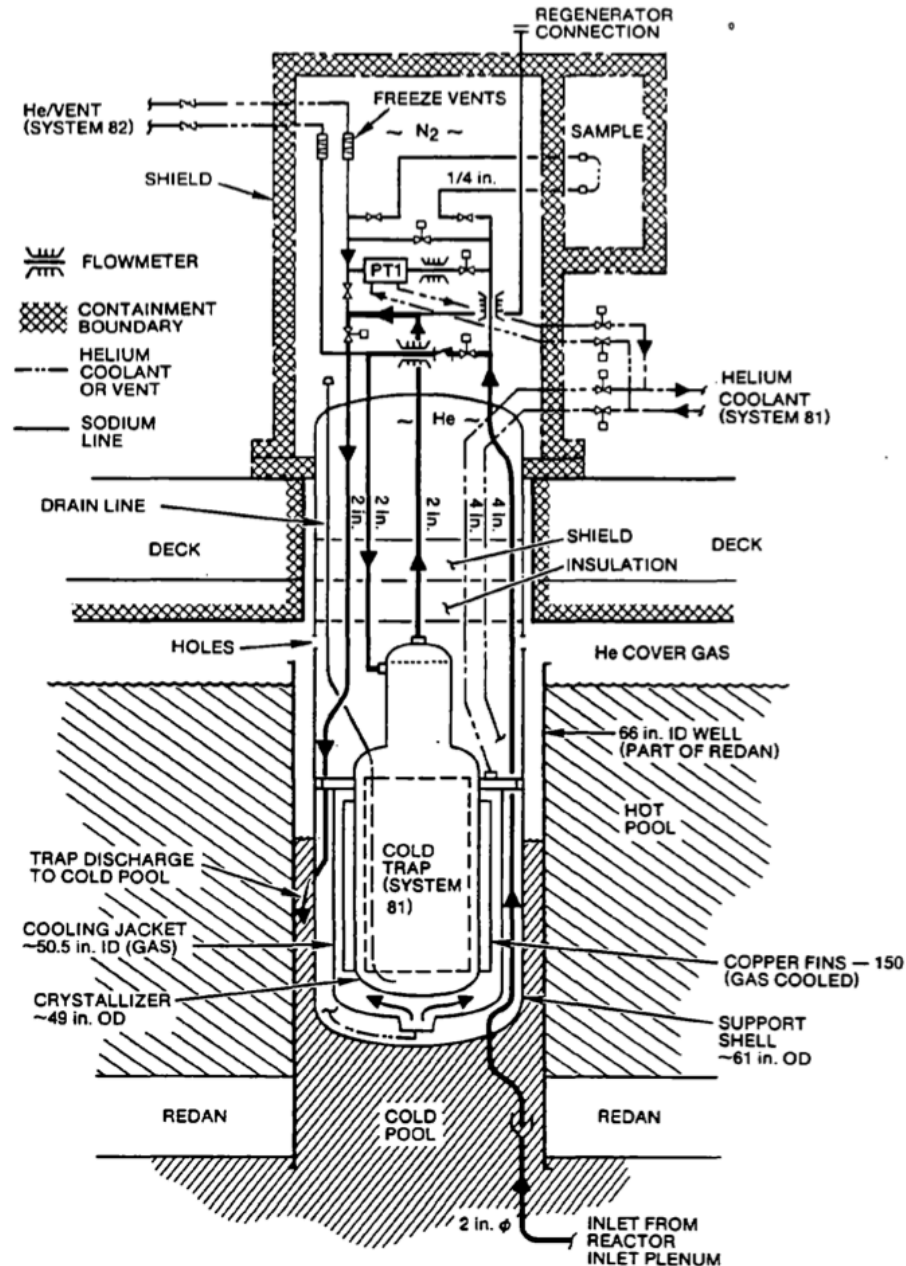
Figure 3–8: SAFR Primary Sodium Purification System Design [22]

Past SFRs have also shown varying capability to capture cesium in the cold trap [41]. For example, in EBR-II it was estimated that approximately 30% of the cesium in the primary circuit was collected by the cold trap after a cladding failure in 1975 [41]. To aid in retention, cesium traps using reticulated vitreous carbon were developed and successfully added to the primary sodium purification loops at EBR-II and FFTF with retention rates at ~90% per pass [41]. At EBR-II, the cold trap did not demonstrate significant retention of iodine.

3.2.3 Cover Gas Radionuclides

> **Cover Gas Radionuclides Summary**
> *The radionuclides of the cover gas region of a pool-type SFR are dominated by ^{23}Ne. However, its short half-life (38s) leaves activated argon (if argon is used as the cover gas) and the noble gases from failed fuel pins (Xe and Kr) as the most active radionuclides.*

There are several sources of radionuclides found in the cover gas region of a pool-type SFR. Sources of varying importance include activation of the cover gas itself, migration of tritium into the cover space, release of decay products from activated sodium, and noble gases from fuel pins that have failed during normal operation.
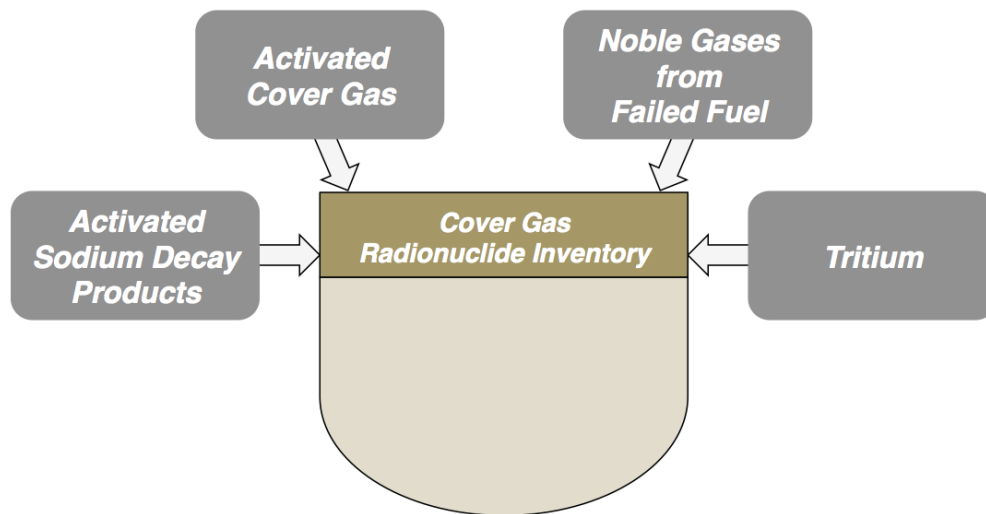
Activated Sodium Decay Products

The largest contributor to the radionuclide inventory in the cover gas region is from ^{23}Ne, which is formed from a (n, p) reaction with ^{23}Na in the primary sodium pool. However, ^{23}Ne has a half-life of only 38 seconds, which means its activity decays away quickly during shutdown.

Activated Cover Gas

The second largest contributor is from activation of the argon cover gas (assuming argon is used as the cover gas). ^{41}Ar can be formed through the following three reactions [41]:

- $(n, gamma)$ reaction with ^{40}Ar in the cover gas region
- $(n, gamma)$ reaction with ^{40}Ar that has become entrained in the primary sodium
- (n, p) reaction with ^{41}K present as an impurity in the sodium

The magnitude of ^{41}Ar activation depends heavily on the reactor design (such as the likelihood of argon entrainment, or the sodium purity levels). ^{37}Ar can also be formed in the cover gas, but has only been found at low levels in past SFRs [41].

Tritium

It is possible for tritium to migrate to the cover gas region, however, the level of activity from tritium is very low when compared to ^{23}Ne, ^{41}Ar, and other fission products [41]. Therefore, tritium is generally not considered a major release source from the cover gas.

Noble Gases from Failed Fuel Pins

If stochastic structural pin failures occur, the noble gases will quickly migrate through the primary sodium, due to their extremely low solubility, and reach the cover gas region. Krypton (^{85m}Kr, ^{88}Kr, ^{87}Kr) and xenon (^{133}Xe, ^{135}Xe) are the dominant radionuclides. The levels found in the cover gas will depend directly on the number of fuel pin failures. It should be noted, that if vented fuel is used, the noble gases released from all fuel pins will reach the cover gas region.

Figure 3–9 shows the maximum recorded activity of the cover gas from FFTF. The activity of ^{23}Ne is several orders of magnitude higher than the next closest source. The activity levels of the noble gases (Xe and Kr) were dependent on the number of fuel failures that occurred during operation.

Isotope	^{23}Ne	^{41}Ar	^{133}Xe	^{135}Xe	^{85m}Kr	^{88}Kr	^{87}Kr	^{3}H
Half-life	38 s	1.8 h	5.2 d	9 h	4.5 h	2.8 h	1.3 h	12 yr
FFTF	74000	15	1900	800	200	200	75	0.0022
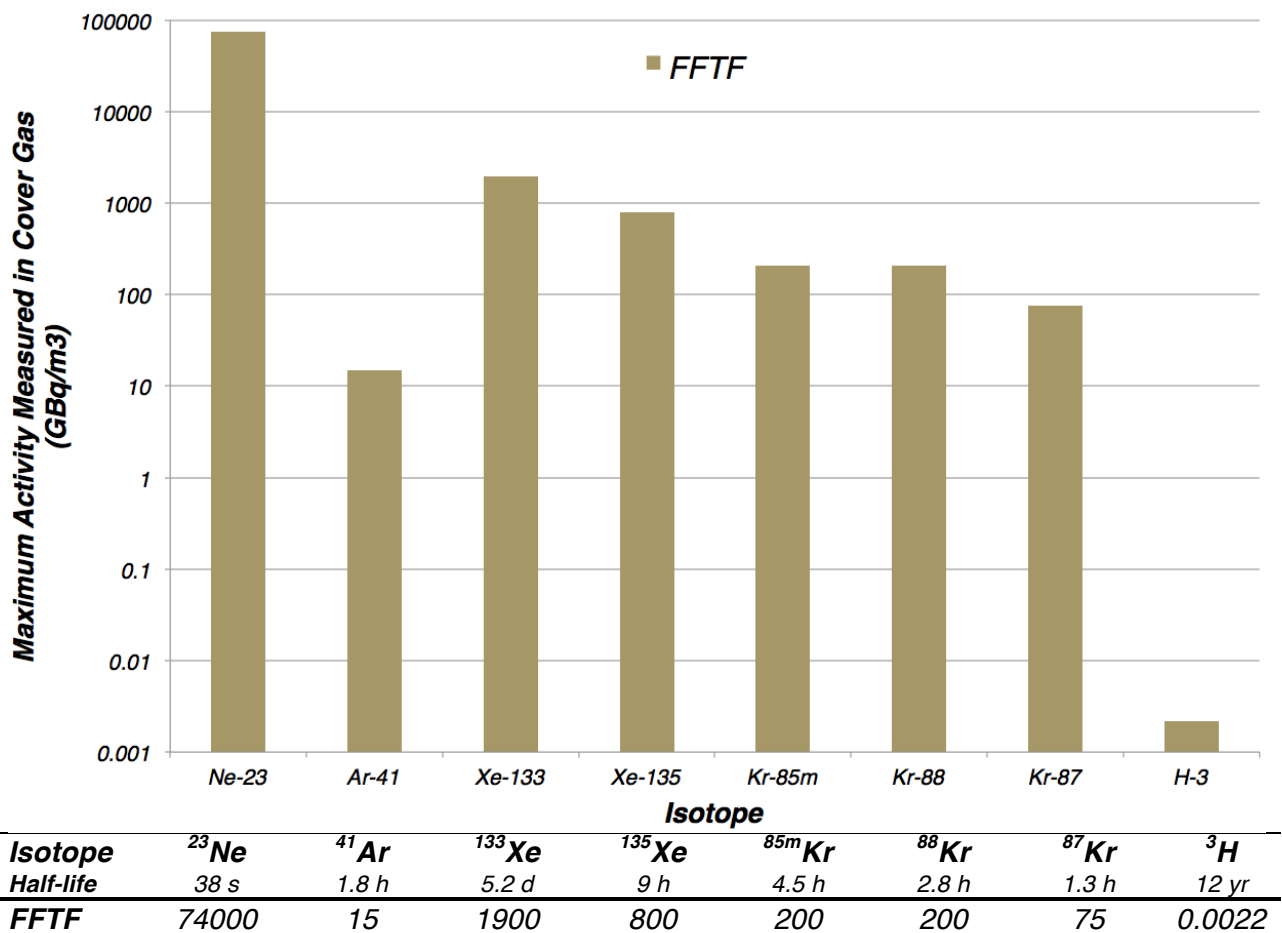
Figure 3–9: Maximum Activity Levels (GBq/m³) Measured in Cover Gas Region [41]

3.2.4 Cover Gas Cleanup Systems

> **Cover Gas Cleanup Summary**
> *Cover gas cleanup systems are not only used to remove impurities from the cover gas region, but can also identify failed fuel pins through gas tagging. For cleanup systems that use a cryogenic distillation column, radioactive fission products are retained in the column, which is placed inside a cold box and usually located outside of containment. The cold box is designed to be capable of retaining the radioactive gas inventory in the event of a leak.*

Cover gas cleanup systems have been used in SFRs not only for the reduction of radionuclides within the cover gas region (which could leak into the reactor containment), but also for the identification of breached fuel pins using an isotope-based gas tagging system.

The cover gas cleanup system at EBR-II, pictured in Figure 3–10, used cryogenic distillation to remove xenon and krypton from the cover gas region. During operation, the argon cover gas and impurities were transferred from the cover gas region and then heated to vaporize any entrained sodium. The gases are then cooled, which allows the vaporized sodium to condense on the surfaces within a retention vessel. From there, the gases leave containment and pass through a heat exchanger and enter a cold box before arriving at the cryogenic distillation column, where condensed xenon and krypton are removed and retained. The clean argon cover gas is then reheated before re-entering the cover gas region. The cryogenic distillation column is located within a cold box, which is designed to hold the distillation column inventory, should a leak occur. Also, the cold box could be vented to an emergency charcoal adsorber. Experience at EBR-II indicates that 99.9% of the xenon and krypton could be removed by the cleanup system [45].

The EBR-II cover gas cleanup system also interfaced with a xenon tag trap. The fuel pins within the core were tagged with unique xenon isotopes. By analyzing the xenon present in the cover gas region, it was possible to identify failed fuel elements.
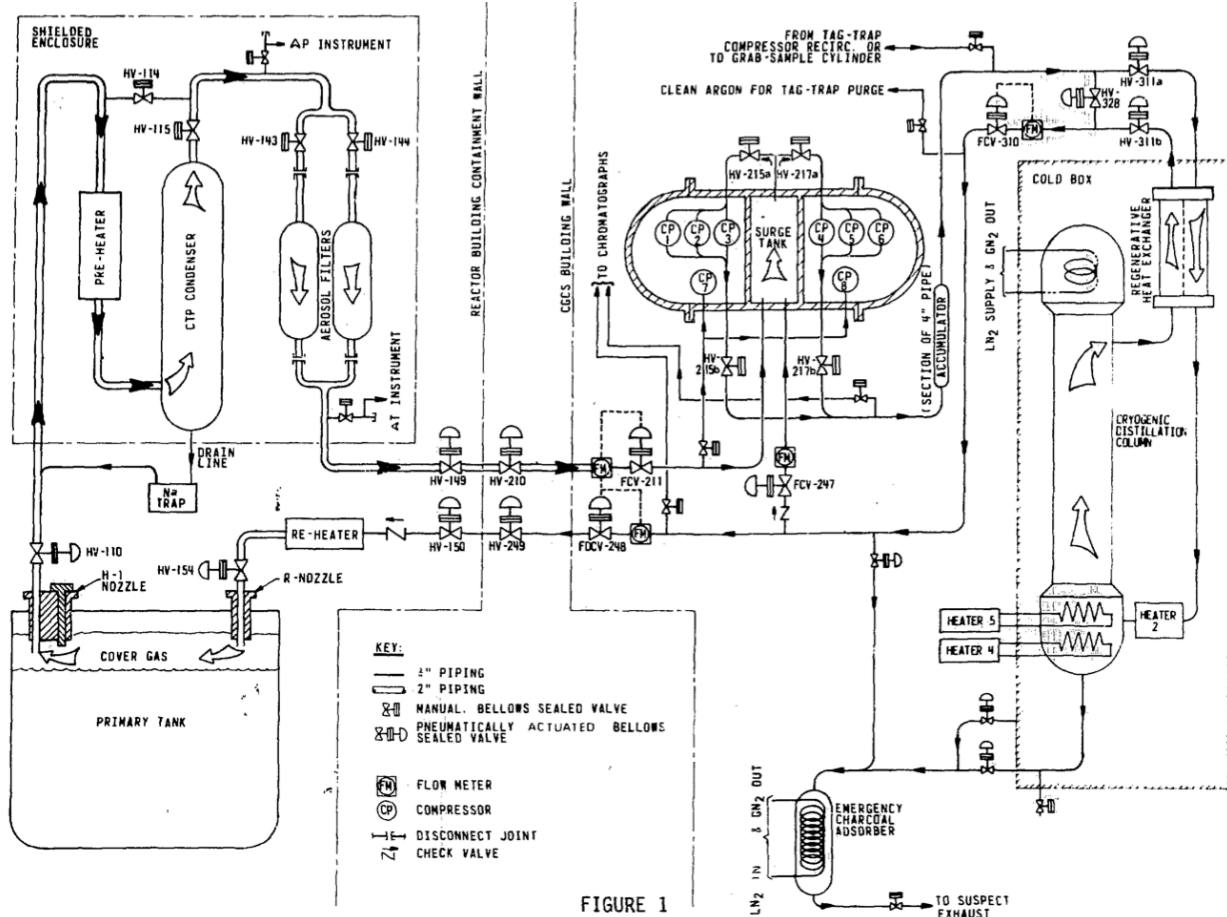
Figure 3–10: EBR-II Cover Gas Cleanup System Diagram [45]

4 Core Damage Source Term Phenomena

The following section examines the phenomena related to radionuclide release from a core damage accident. This begins with a review of the phenomena associated with radionuclide release and transport, and is followed by an examination of the current state of knowledge and an assessment of past accidents, experiments, and analyses. It should be noted that past efforts to review SFR source term phenomena, such as ref [46], were vital to the current effort.

While the following sections do not examine a specific core damage incident, it restricts the phenomena reviewed to those associated with cladding breach and fuel melting with limited propagation, and assumed in-vessel retention of molten fuel. This choice was based principally on the preliminary findings of the PRISM and SAFR[12] PSIDs [22] [24], but was also impacted by the experiences gained from the EBR-II, Fermi 1 and SRE fuel melt accidents (discussed in Section 4.3).

Beginning with the CRBR licensing process and continuing with the PRISM and SAFR NRC interactions, SFR accidents have typically been identified as either design basis protected accidents (*i.e.*, with scram) or *beyond* design basis unprotected accidents (*i.e.*, without scram). For both the PRISM and SAFR reactor designs, the PSID analyses indicated that fuel damage would not occur for any design basis accident. The beyond design basis accidents were also analyzed as part of a bounding assessment, and the findings indicated that fuel damage was unlikely to occur for what was considered the "base" beyond design basis accidents (*i.e.*, an unprotected design basis accident scenario). Only when additional faults were included in the analysis (*i.e.*, an unprotected design basis accident with additional failures) was minor core damage[13] found to occur [22] [24].

While the PRISM and SAFR PSID results are not meant to be representative of all metal-fuel pool-type SFR designs, similar core damage results are usually taken as general design requirements by SFR vendors. That is to say, design basis accidents should not result in core damage, while the most credible (probabilistically) beyond design basis accidents should result in little to no core damage. For beyond design basis accidents that lead to minor core damage, in-vessel retention and coolable geometry should be ensured.

Therefore, the core damage phenomena reviewed in this section is considered bounding for the most probable core damage scenarios, even though they are likely to fall outside of the design basis accident space. Those accidents that are likely to be at or below the residual risk range in probability for metal-fuel pool-type SFR analyses (such as HCDAs, significant core damage coupled with large primary sodium fires (loss of reactor head), and vessel melt-through) are not considered. As described in Section 2.3.2, part of the motivation of the transition to metal-fuel pool-type SFR designs after the CRBR project was the ability to leverage inherent reactivity feedback mechanisms, high thermal conductivity, and low in-fuel heat retention as tools to prevent and limit such core damage events.

---

[12] Both the PRISM and SAFR designs are metal-fuel pool-type SFRs.

[13] While there is no consensus definition of core damage for a metal-fuel SFR, here, core damage is taken as cladding failure of multiple fuel pins.

4.1 Transport Phenomena

For a metal fuel SFR, there are five main barriers to the transport of radionuclides:

1) **Fuel Matrix** – Retention of radionuclides within the fuel matrix
2) **Cladding** – Retention of radionuclides within the fuel pin
3) **Primary Sodium** – Retention of radionuclides within the primary sodium
4) **Primary Circuit Boundary** – Retention of radionuclides within the primary system (includes the primary sodium and cover gas region)
5) **Containment** – Retention of radionuclides within the containment building and guard vessel

Sections 4.1.1 through 4.1.4 step through each of these barriers and identify the phenomena that will likely be encountered by both radionuclide vapors/gases and particulates. Only the high-level phenomena are reviewed in this section. Sections 4.2 and 4.3 will discuss the phenomena related to particular radionuclide groups while reviewing the current state of knowledge and past accidents. The focus here is on the transport and retention of radionuclides released from the fuel. Phenomena related to the melting, relocation, freezing, and retention of fuel within the core are not discussed here, but ref [47] provides a detailed review of these issues.

The following two assumptions are made for the phenomena reviewed in this report:

- Possible retention of radionuclides within the primary sodium purification or cover gas cleanup system is not considered, as these systems are designed to support normal operation and are usually not intended to cope with accident conditions and may not be safety grade.
- Radionuclide transfer to the intermediate loop (through failure of the intermediate heat exchanger in the sodium pool) is not considered since the intermediate loop is at a higher pressure and elevation than the primary sodium. Therefore, any breaches in the intermediate heat exchanger should result in leakage from the intermediate loop into the primary sodium, rather than sodium leakage from the primary system into the intermediate loop.

A brief review of relevant chemical phenomena will aid in comprehension of the following sections. First, the nomenclature shown in Figure 4–1 is used to represent the phase transitions of elements and compounds. This is important as radionuclides may change phases as they are released from the hot fuel pin and transport to relatively cooler regions of the primary sodium or cover gas region. The term *vapor* is used for a substance in the gas phase but at a temperature lower than its critical point.
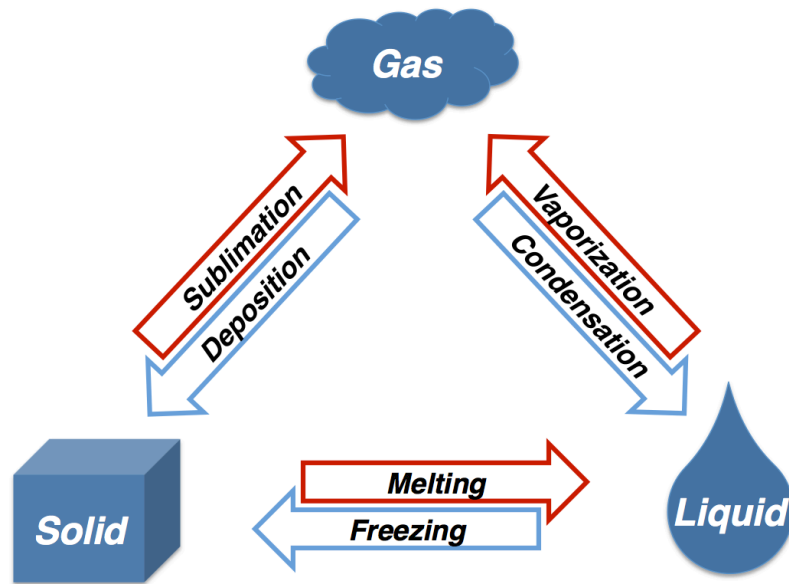
Figure 4–1: Phase Change Terminology

In addition, the following phenomena occur at different stages of radionuclide transport:

Mechanical Deposition - Used to differentiate the deposition of particles onto surfaces due to gravitational settling, impaction/interference, diffusion, and thermophoresis from the phase change of deposition.

Adsorption - Used interchangeably with *plateout* to refer to the adhesion of elements/compounds from a liquid or gas onto a surface. It is different from mechanical deposition since it implies a bonding between the surface and the adsorbed liquid or gas (by both chemisorption and physisorption).

4.1.1 Retention within the Fuel Pin
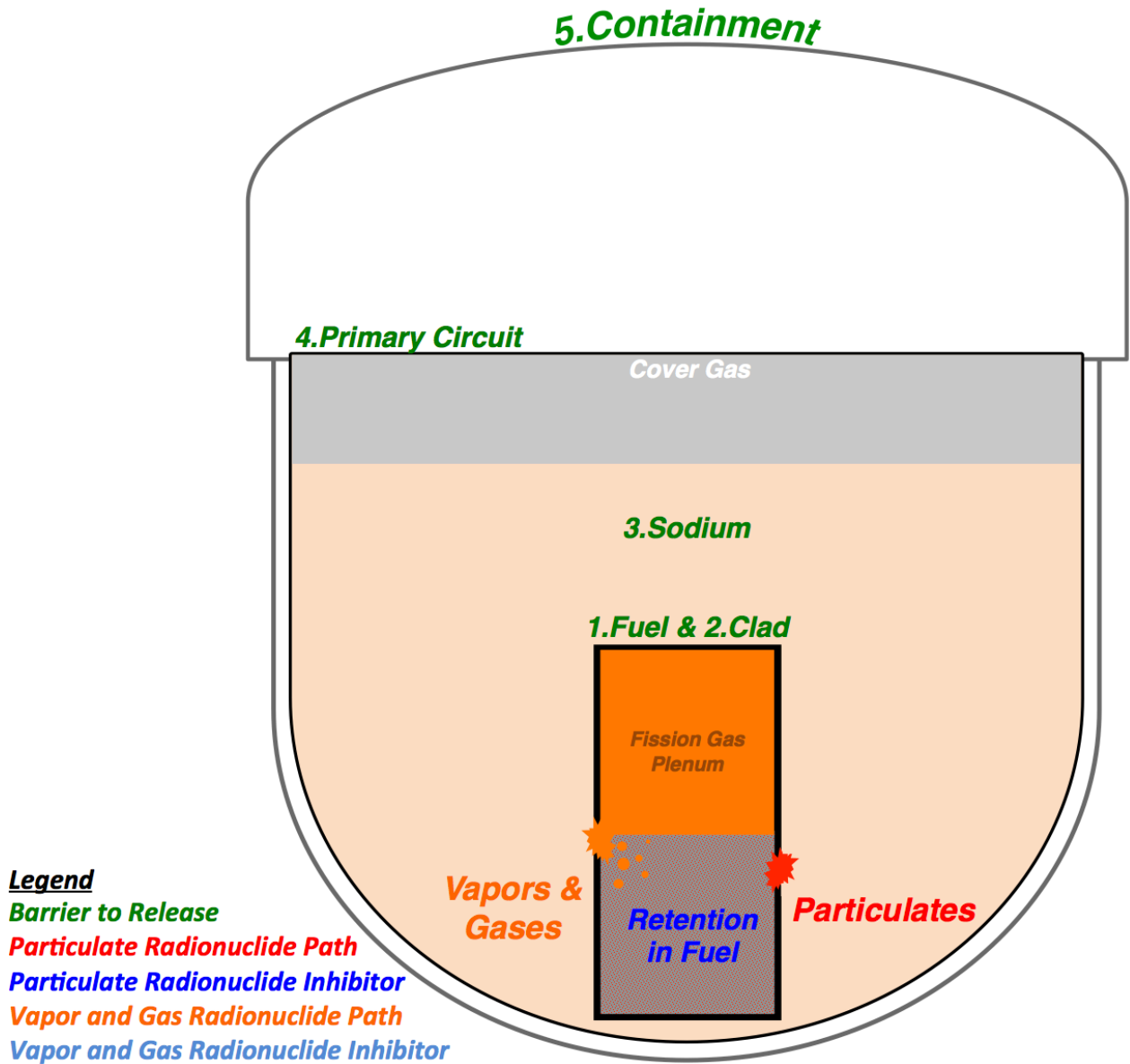
Figure 4–2: Core Damage Accident – Retention in Fuel Pin Phenomena

As fuel burnup increases during normal operation, the creation of fission gases causes the fuel to swell and come into contact with the cladding. The fission gas pockets will interconnect and form passageways to the fission gas plenum, which in turn significantly slows further fuel swelling. Some fission products may migrate to the bond sodium, which is displaced to the fission gas plenum due to fuel swelling. Fission gases and vapors released from the fuel matrix must first travel through the bond sodium before reaching the fission gas plenum.

Elevated temperatures during an accident will cause eutectic penetration of the cladding to begin, with eutectic thinning rates of the cladding dependent on fuel temperatures and thermally-induced cumulative cladding stress. If fuel temperatures remain around the eutectic formation temperature for an extended period of time, cladding breach may occur. However, to encounter fuel melting in

areas of the fuel other than the eutectic regions, higher fuel temperatures would be required. If an accident causes extreme fuel pin temperatures with significant internal fuel melting, more rapid eutectic penetration and cladding failure is expected.

With a breach in cladding, the radionuclide gases and vapors contained in the fission gas plenum will be released to the primary sodium, along with the bond sodium (and any fission products that have migrated to and dissolved in the bond sodium). If the fuel pin failed due to eutectic cladding penetration, only the outermost eutectic region of the fuel will be molten. Without significant additional fuel melting, there is likely to be minimal release from the fuel, due to the compatibility (*i.e.*, non-reactivity) between metal fuel and sodium, as will be described in Section 4.3.4.

Many radionuclides will be retained in the fuel matrix, as uranium acts as an excellent solvent. If additional fuel melting were to occur, due to a prolonged loss of cooling or a power spike that results in higher fuel temperatures, fuel material, including fission products retained in the fuel, may enter the coolant channel. This will result in a greater release of radionuclides into the primary sodium. However, once again, due to the compatibility of metal fuel and sodium, there is no chemical reaction between the two substances.

4.1.2 Retention within the Primary Sodium
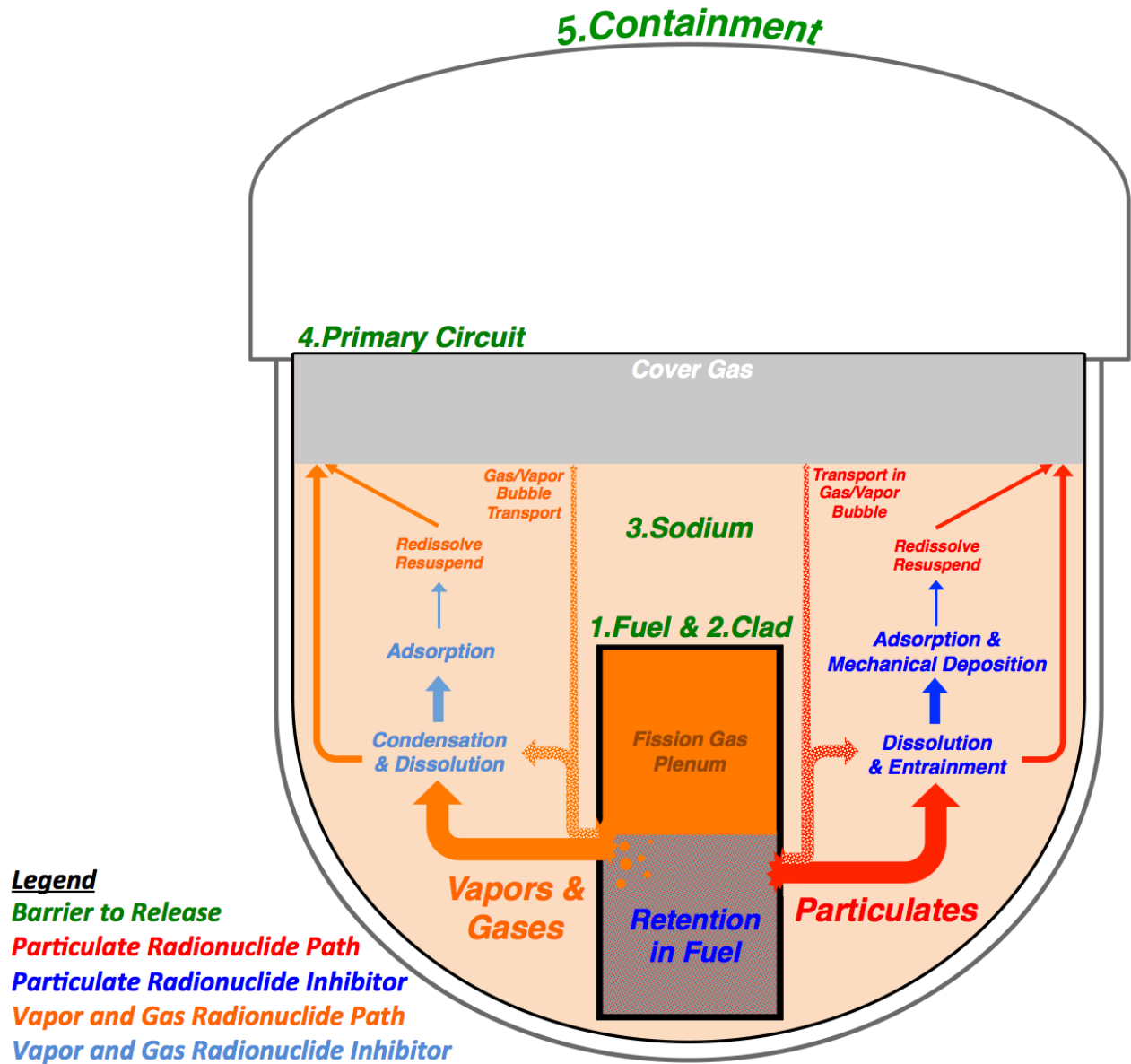
Figure 4–3: Core Damage Accident – Retention in Primary Sodium Phenomena

Once released from the fuel pin, the behavior of radionuclides within the primary sodium is complex. There are many chemical thermodynamic considerations that must be taken into account to predict the retention of the radionuclides in the sodium. Appendix A provides a more detailed discussion of the factors that influence radionuclide mixing in the primary sodium. This section provides a brief overview of the phenomena that will likely affect the behavior of radionuclide vapors/gases and particulates.

Vapors and gases released from the fuel pin have differing fates depending on the vapor pressure and solubility of the particular element or compound in the sodium solution. Gases or vapors with high vapor pressure and low sodium solubility (like the noble gases discussed in Section 4.2.1) will be directly transported through the primary sodium and to the cover gas region. Some vapors

may condense completely to the liquid phase once they come in contact with colder sodium and could dissolve, or they may nucleate within a bubble and be transported as aerosols. Other vapors will directly dissolve in the primary sodium from the gas phase due to high solubility of the element or compound in sodium. Adsorption of the dissolved vapors onto primary system structure may also transpire, but this phenomenon is dependent on properties of the individual element or compound and the material of the primary system structures.

Particulates that are released from the fuel pin may become entrapped within a vapor or gas bubble. These particulates may settle on or migrate to the surface of the bubble and interact with the sodium; the same outcome will occur if the bubble collapses as the vapor condenses when colder sodium is encountered. The particulates may dissolve in the primary sodium or become entrained in the moving sodium stream. The particular phenomenon that will be encountered by the particulate depends on whether compounds are formed (*i.e.*, chemical reactions with the sodium) and the solubility of the element/compound in sodium. From there, adsorption on structure may occur, especially in lower temperature regions of the primary system where dissolved radionuclides may precipitate. Mechanical deposition within the primary system is also a possibility, especially for entrained particles.

For those radionuclides that have adsorbed onto structure, redissolution is possible if temperature changes occur in the primary sodium, as solubilities are typically highly dependent on temperature. If a radionuclide has mechanically deposited onto structure, resuspension may be possible is flow conditions adjacent to the structure change.

4.1.3 Behavior in Cover Gas
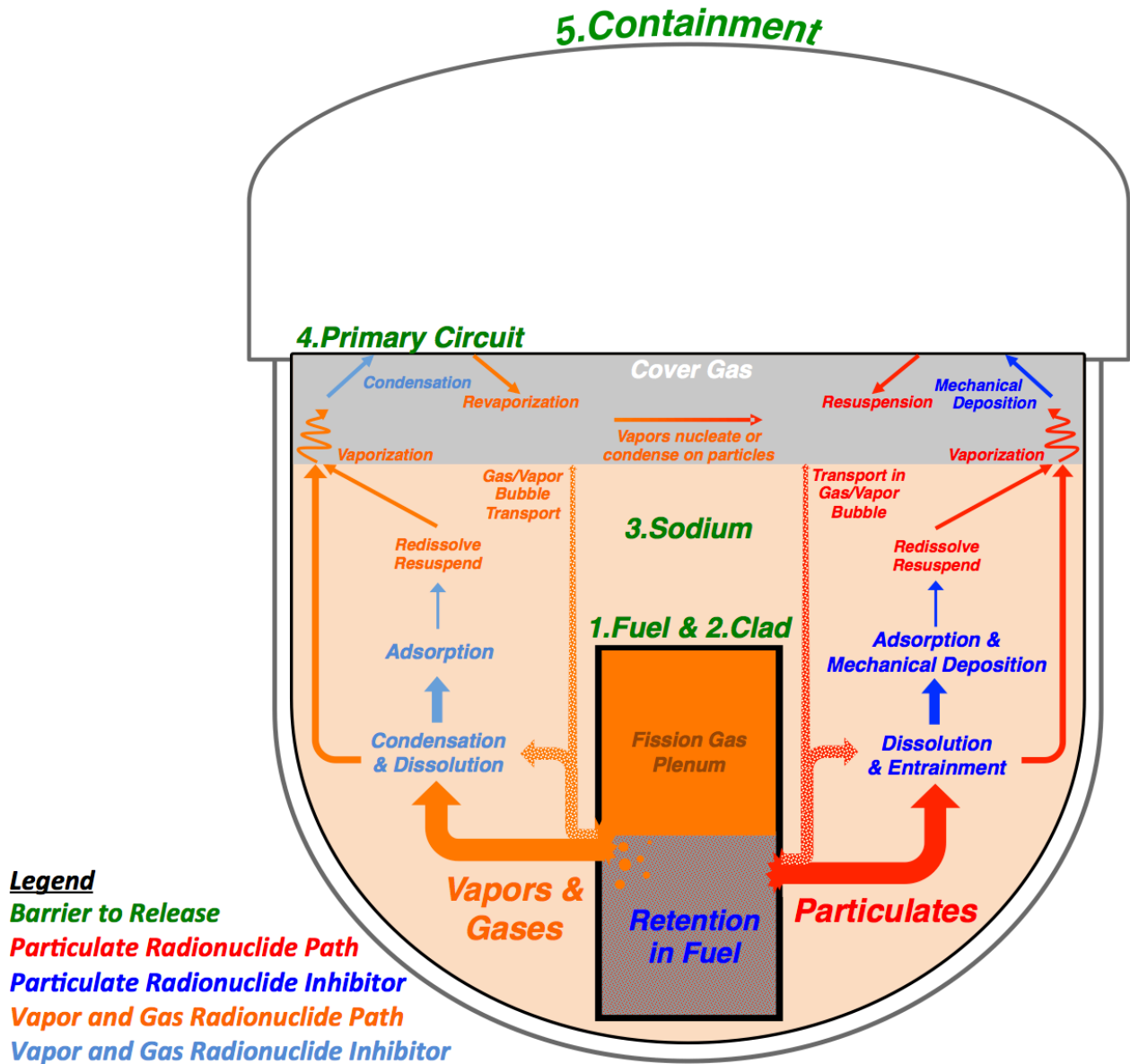
Figure 4–4: Core Damage Accident – Behavior in Cover Gas Phenomena

Those vapors and gases that were directly transported through the primary sodium in bubbles due to their high vapor pressure and low solubility (such as the noble gases) will be released to the cover gas region upon reaching the sodium surface. If any entrained particles are present within these bubbles, they may also be initially released to the cover gas region when the bubble bursts at the surface of the sodium pool.

The vapors and gases that dissolved in the primary sodium, along with dissolved particulates, must vaporize in order to escape the sodium. Barring any significant sodium boiling event, this can only occur through evaporation. The vapor pressure of a substance determines its ability to evaporate from a liquid. Mixing in the sodium pool can greatly affect a substance's vapor pressure; Appendix A describes the thermodynamic factors that influence this phenomenon.

The radionuclides that vaporize from the pool and reach the cover gas region will encounter a temperature decrease. This can lead to the condensation of vapors on structural surfaces or onto particles. Mechanical deposition can again remove particulates from the cover gas region, as particles impact the various structures present, or settle back onto the sodium pool due to gravity. As will be described in Section 4.2.1, for highly volatile vapors or gases, such as the noble gases, condensation will not occur even in this cooler environment.

Lastly, resuspension/revaporization of some radionuclides is again possible if temperature changes occur in the cover gas region, as volatility typically increases with increasing temperature. Mechanically deposited particles could resuspend if a mechanical shock occurs on the structure where the particles are located, or if vapor flow increases adjacent to the structure.

As will be discussed in Sections 4.2 and 4.3, there is significant radionuclide retention in the primary sodium for all radionuclide groups other than the noble gases. Subsequent vaporization to the cover gas region is very small, and requires an extended period of time to occur. These phenomena, along with radionuclide retention within the fuel matrix, are important as they can significantly reduce the radionuclide inventory that is available for release from subsequent barriers in the defense-in-depth philosophy.

4.1.4 Behavior in Containment
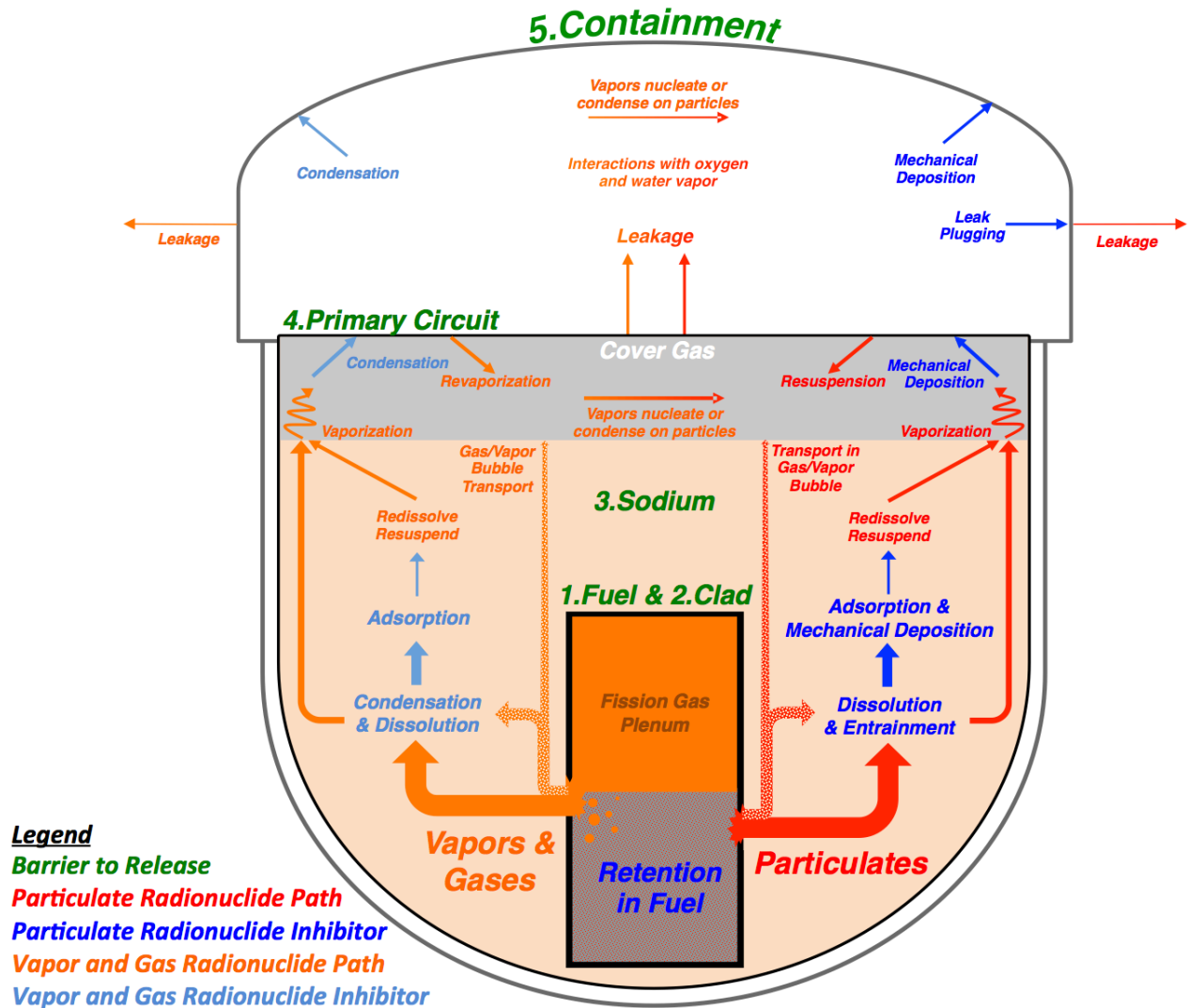
Figure 4–5: Core Damage Accident – Behavior in Containment Phenomena

Airborne radionuclides may transport to the upper containment area through leakage paths in the reactor head. Typically, the reactor head has a design basis leak rate, which is very small (on the order of 0.1% volume per day)[14]. Once in containment, some of the same phenomena associated with radionuclide transport in the cover gas region will occur, while additional phenomena are possible due to the presence of oxygen and water vapor.

First, much colder temperatures than the cover gas region will likely be found in containment, which will encourage additional condensation of vapors either onto the surface of structures, or

---

[14] The integrity of the reactor vessel head and associated penetrations is an important factor in SFR MST calculations, as retention of vaporized radionuclides in the cover gas region can greatly reduce the release into containment, and subsequently the environment. A large radionuclide release to the cover gas region may cause a temperature increase and a pressure rise beyond the maximum operating pressure of the reactor head, which could result in increased leakage, beyond the design basis.

onto aerosols (although the highly volatile noble gases will remain in the gas phase). Mechanical deposition will once again reduce any particles that managed to successfully transport from the cover gas region into containment.

Unlike the primary sodium or cover gas region, there is likely to be oxygen and water vapor in the containment, as it is typically not inerted to allow for maintenance activities. Leakage from the cover gas region means that sodium vapor in the cover gas region may also enter containment (the concentration of sodium vapor in the cover gas region is likely to be much higher than the concentration of radionuclides). The sodium vapor will react with the oxygen and water vapor to form aerosol particles. These particles will agglomerate and may remove vapors or particulates that have condensed or mechanically deposited on their surfaces. Other radionuclides may also react with the oxygen and water vapor and decompose to form new compounds.

The containment is expected to have a small design basis leak rate, similar to that of LWRs, which will permit some release of the radionuclides that have managed to remain airborne. The pressure within containment is unlikely to be elevated much beyond normal conditions (ambient atmospheric conditions), barring some other associated accident (such as sodium fire associated with the intermediate sodium piping)[15]. Unlike LWRs, the primary circuit of an SFR is near atmospheric pressure, meaning any failures in the primary system barrier do not result in the release of significant amounts of energy to containment. Leak plugging is another interesting phenomenon that may limit leakage from the containment. The aerosols formed in containment due to condensation and reactions with oxygen/water vapor may plug leakage pathways. As airflow through these leaks carries the aerosols to them, they can mechanically deposit around and in the leak causing a reduction in flow area.

---

[15] It may be possible that containment temperatures rise due to exothermic reactions between the oxygen/water vapor and the materials released from the cover gas region. However, this would be highly dependent on the magnitude and form of materials released from the cover gas region, the size of the containment, and the quality of the air in containment.

4.2 Current State of Knowledge

The following section reviews the current state of knowledge regarding the phenomena described in the preceding analysis of radionuclide transport for a core damage accident. The radionuclides are separated into representative categories, as is done in TID-14844 and NUREG-1465. However, the dominant radionuclide groups and isotopes may vary from what have historically been important for LWR source term considerations. In particular, in the past, isotopes identified as key to the source term analysis have generally been those that are considered the most important to potential human radiation dose. Consequences of the Great East Japan Earthquake in 2011 have led to the recognition that the loss of habitable land and food production are also important to source term assessments, although there is no current regulation regarding these factors. While efforts were taken to review the full spectrum of possible radionuclides, the focus of the following subsections remains on those elements and isotopes that have historically been deemed important for human health considerations.

For each radionuclide group, the knowledge regarding the following factors is assessed:

- Behavior in the fuel pin
- Behavior in sodium pool (solubility, mixing)
- Possible release from the sodium pool (vaporization)
- Aerosol behavior (in cover gas region and in containment)

Not every factor is reviewed for each radionuclide group, as some radionuclides will likely be retained by one of the earlier occurring mechanisms, and are unlikely to vaporize and reach the aerosol state. The current state of knowledge focuses on experimental results and theoretical models for the behavior of the radionuclide groups. How these radionuclides behaved in past accidents is described in more detail in Section 4.3.

4.2.1 Noble Gases

> **Noble Gases Summary**
> *The noble gases of interest, xenon and krypton, form as small bubbles in the fuel matrix. With increasing burnup, the bubbles agglomerate and migrate to the fission gas plenum through interconnected porosity. Any subsequent cladding breach will likely result in a significant release of the noble gas inventory to the primary sodium. From there, both noble gases will quickly (within minutes) migrate to the cover gas due to their negligible solubility in sodium and high vapor pressure. As fixed gases, there will be no condensation of xenon or krypton in the cover gas region or containment. However, the time delay of release (assuming a design basis leak rate from the cover gas and containment) can significantly reduce the total radioactive inventory.*

The noble gases of interest, xenon and krypton, are formed from both the direct fission of uranium and through the decay of other fission products in the fuel matrix. During operation, fission gas bubbles containing both xenon and krypton are created, causing the fuel to swell and internal pin pressure to rise as the burnup level increases. The fuel porosities begin to

interconnect, forming pathways out of the fuel matrix, which allow the fission gas bubbles to travel through these new pathways and eventually reach the fission gas plenum [35].

Any breach of the cladding is likely to cause the bond sodium to escape from the fuel pin and result in a pathway from the fission gas plenum to the primary sodium. As shown in Figure 4–6, fission gas release to the plenum (including xenon and krypton) increases substantially beginning with a fuel volume increase due to swelling of ~30%.
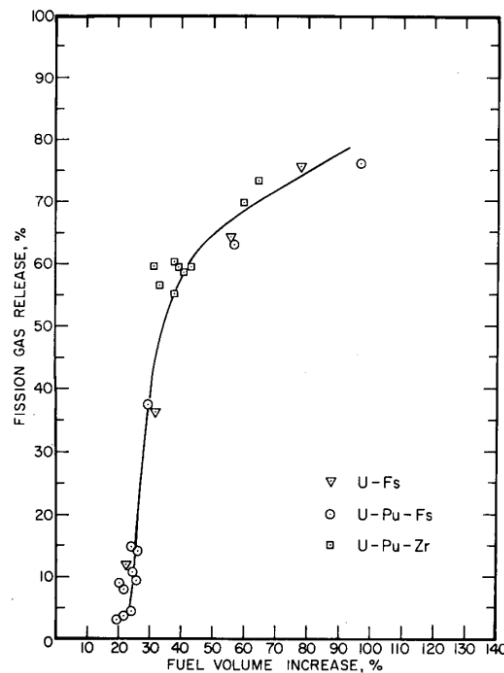
Figure 4–6: Fission Gas Release versus Fuel Volume Increase [48]

The magnitude of volume increase is correlated with the burnup level, but the exact composition and smear density of the fuel will also have an effect. However, as Figure 4–7 shows, it is likely that the majority of the xenon and krypton formed in the fuel pin will be released from the fuel pin during a cladding breach if burnup levels are beyond a few percent [48] [49].
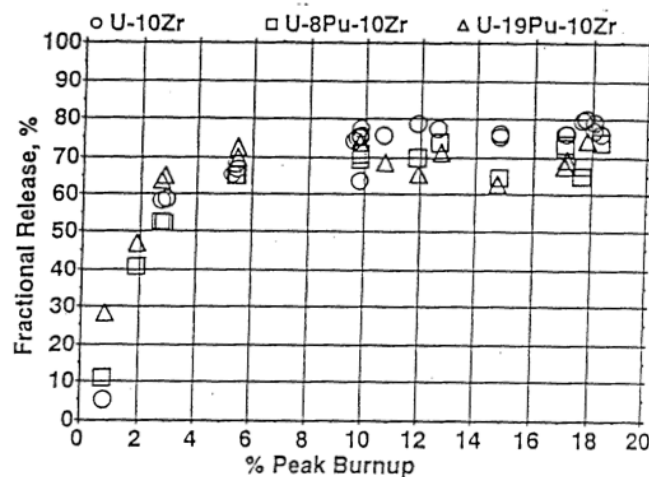
Figure 4–7: Fission Gas Fractional Release Burnup Dependence [49]

Once xenon and krypton enter the primary sodium, their solubility obeys Henry's law [50]. Multiple data sources have demonstrated that the retention of xenon and krypton in sodium under conditions of the primary system is negligible [50] [51], due to a high vapor pressure and a low solubility that is about a factor of 10^4 smaller than in water. Therefore it can be assumed that any xenon and krypton released into the primary sodium will reach the cover gas region. The time delay between release from the fuel pin and entrance to the cover gas will vary with reactor design, but has been estimated to be on the order of several minutes for previously constructed reactors [52][16].

Condensation of xenon and krypton in the cover gas region will not occur due to their high vapor pressures. Some xenon and krypton will escape the cover gas region, assuming a design basis leak rate from the cover gas region into containment. Once in containment, the noble gases will mix with the existing gas in the containment and escape to the environment according to the containment design basis leak rate.

It is important to account for the time delays that occur during the release of noble gases from the fuel to the cover gas region and into containment. Many of the radioactive isotopes of xenon and krypton have relatively short half-lives, as seen in Table 4-1, which also shows the dominant decay mode and corresponding energy. This delay provides time for a reduction in the xenon and krypton radionuclide inventory. While transport within the primary sodium may only take several minutes, the retention of xenon and krypton in the cover gas region and within containment may be much longer, assuming design basis leak rates. Figure 4–8 shows how the activity from xenon and krypton for a single SFR metal-fuel assembly would decay with time.

Table 4-1: Major Radioactive Isotopes of Xenon and Krypton

Element	Isotope	Half-Life	Decay Mode	Energy (keV)
Xe	133	5.2 d	β^-	427.36
	135	9.1 h	β^-	1165.0
	137	3.8 m	β^-	4162.4
	138	14.1 m	β^-	2914.8
Kr	85	10.7 y	β^-	687.00
	87	1.3 h	β^-	3888.3
	88	2.8 h	β^-	2917.7

---

[16] The travel time will be affected by bubble size, pool depth, presence of structures, and swarm effects.
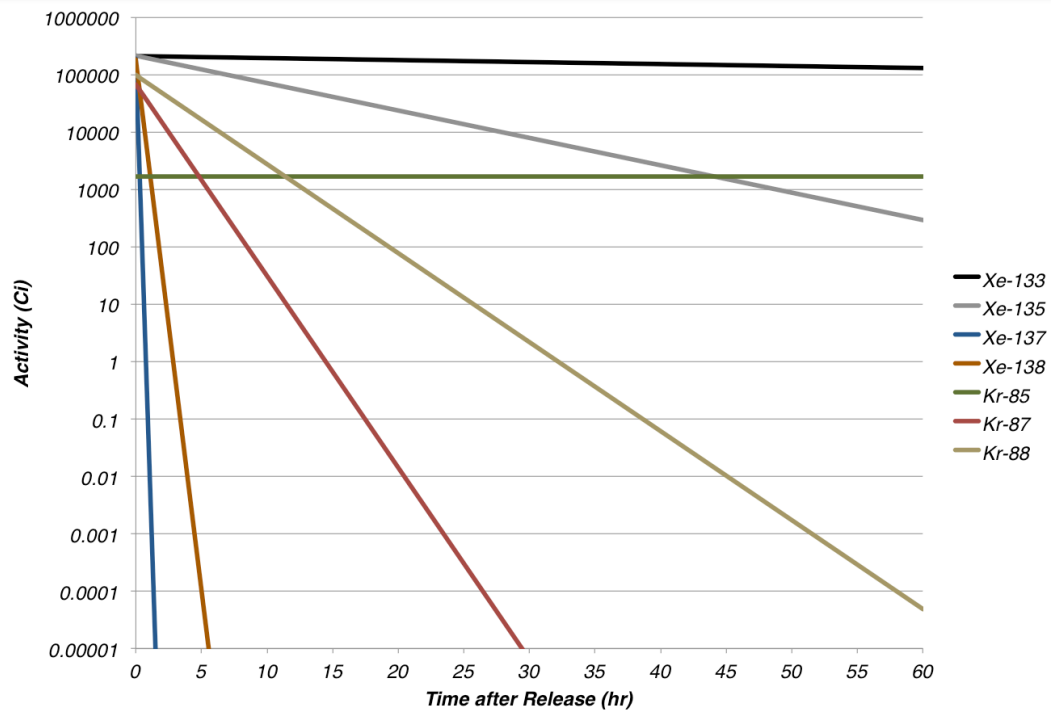
Figure 4–8: Noble Gases Radioactivity Decay with Time[17]

4.2.2 Halogens

> **Halogens Summary**
> *Iodine is the main halogen of interest to MST development, since bromine has a short half-life and low fission yield. Within the fuel, the formation of UI₃ and CsI is likely. The iodine within UI₃ can only be released if significant fuel melting occurs. The CsI may migrate to the bond sodium and be released if the cladding is breached. In the bond or primary sodium, CsI decomposes to Cs and NaI, which is a very stable iodide. The NaI may collect at the gas-liquid boundary of the sodium pool, but significant NaI vaporization is unlikely due to its low vapor pressure (the vaporization of NaI lags the vaporization of sodium). This also means any vaporized NaI would likely quickly condense on cooler surfaces in the cover gas region.*

Iodine and bromine are the halogens usually assessed during source term development. However, due to their low fission-yield and short half-lives, isotopes of bromine are likely not a major contributor to the radiological source term. Iodine, however, is of interest and its form and behavior in metal fuel differs greatly from what is found in oxide fuels, such as those used in LWRs.

There are four isotopes of iodine that are likely to be formed in significant quantities in the fuel, which are shown in Table 4-2; only the dominant decay modes and corresponding energy are shown for each isotope.

---

[17] Inventory is for one assembly of a typical metal fuel SFR at end-of-cycle. Release from fuel pins assumed to occur at time zero.

Table 4-2: Dominant Isotopes of Iodine

Element	Isotope	Half-Life	Decay Mode	Energy (keV)
I	127	Stable	-	-
	129	1.57×10^7 yr	β^-	188.92
	131	8.0 d	β^-	970.85
	133	20.8 h	β^-	1757.0

For radiological hazard reasons, the ^{131}I isotope is of primary interest in the development of an MST; while all isotopes of iodine collect in the thyroid, the longer half-lives combined with a relatively high fractional contribution to the total core inventory make ^{131}I and [133]I important MST contributors from the iodine group. They are formed almost entirely from the decay of fission products within the fuel [53]. The presence of elemental I_2 in the fuel is very unlikely since iodine will bond with other elements [54], and gaseous I_2 has not been seen in past reactors. In order of thermodynamic preference, iodine will form the following iodides: CsI, BaI_2, SrI_2, LaI_3, CeI_3, NdI_3, and UI_3.

Even though UI_3 is not the thermodynamically preferred iodide, the ratio of uranium to the other possible iodide elements within the fuel is so high (several orders of magnitude greater) that UI_3 is still likely to form and be a major factor in the retention of iodine within the fuel [53]. CsI is also likely to be found since cesium is produced at a faster rate than iodine and has thermodynamic preference compared to other fission products [54].

As shown in Figure 4–9, CsI may migrate to and collect in the bond sodium of the fuel pin, and during a cladding breach, may be released into the primary sodium. In sodium, CsI will decompose to form Cs and NaI [51]. NaI is a very stable compound of iodine (only the alkali salts, such as CsI, and akali earth salts, such as BaI_2, are more stable [55]).
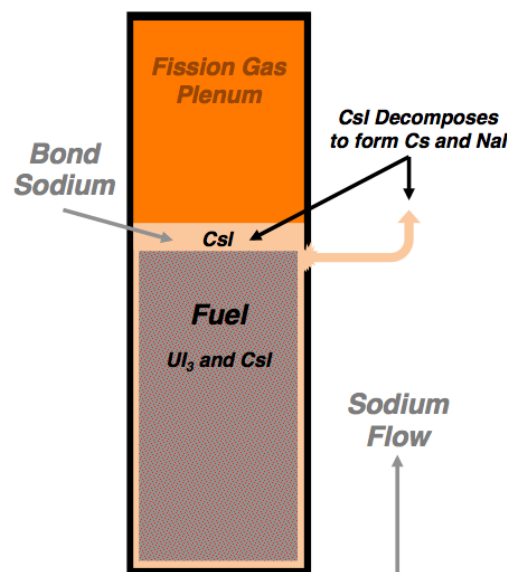
Figure 4–9: Movement of CsI from Fuel Pin

Uranium, and therefore UI_3, will not be released from the fuel in any appreciable quantity due to a cladding breach. Only if fuel melting is encountered can this iodine be released into the primary

sodium to form NaI [51] [56]. Experimentation with melting of an U-Fs alloy in a crucible for four hours at 1250°C resulted in the release of only ~1% of the iodine from the fuel[18] [57]. Only when held at higher temperatures (≥1300°C) for multiple hours were significant fractions of the iodine released from the fuel [57].

Once in the primary sodium, iodine will remain as NaI and not recombine with cesium [56]. Past experiments have shown that NaI will collect near a gas-liquid boundary, rather than disperse as a homogenous mixture [51]. This observation may influence vaporization rates from the sodium pool surface. Precipitation is unlikely to occur since the iodine concentration will likely be very low, and NaI stays in the liquid phase at lower temperatures when highly diluted in sodium [55] [58].

NaI has a very low vapor pressure [59], and multiple studies and experiments have examined the vaporization of NaI [60] [61] [62] [63]. Equilibrium vaporization results[19] from [60] are shown in Figure 4–10. At typical primary sodium temperatures (~800 K) NaI vaporization lags the vaporization of sodium as a whole (*i.e.,* more than 50% of sodium needs to vaporize for approximately 10% of NaI to vaporize). The comparison to sodium vaporization provides an intuitive measure to gauge the vaporization of a radionuclide. Since sodium vaporization from the pool is usually very small (the fraction released to the cover gas is usually on the order of 10^{-4} - 10^{-6}), and NaI vaporization lags sodium, it gives an indication of the very small release fraction of NaI. Barring a significant sodium vaporization event, such as very high pool temperatures or bulk pool boiling, it is unlikely that significant quantities of NaI will vaporize from the pool [54].
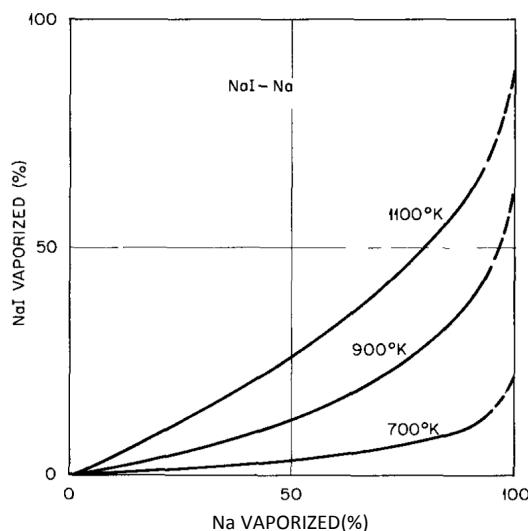
Figure 4–10: Vaporization of NaI [60]

Even if NaI were to vaporize, its very low vapor pressure would likely result in its condensation on cooler surfaces in the cover gas region [59]. If any vaporized NaI were able to reach the

---

[18] 1250°C is above the melting point of most SFR metal fuels (1000-1200°C). Eutectic formation occurs at even lower temperatures.

[19] These vaporization curves assume equilibrium and were constructed based on the results of experiments to determine the excess Gibbs free energy. More detail on how this information can be translated to vaporization data can be found in Appendix A.

containment atmosphere before condensing, it could react with water vapor in the air to form molecular iodine [46].

4.2.3 Alkali Metals

Alkali Metal Summary
Cesium is the alkali metal of most importance to MST development. Within the fuel, some of the cesium will form CsI, but once the CsI collects in the bond sodium, it will decompose to Cs and dissolve in its sister alkali metal sodium. A cladding breach will release the bond sodium, including the dissolved cesium, to the primary sodium. Cesium retention in the sodium is expected to be very high, with little adsorption occurring. While cesium is more volatile than sodium or NaI, it is unlikely that significant amounts of cesium would be released to the cover gas. Any vaporized cesium will likely condense on colder surfaces in the cover gas region.

Cesium and rubidium (Rb) are the alkali metals meriting strongest consideration for source term development, with the focus on the release and transport of cesium. The cesium isotopes formed in significant quantities are shown Table 4-3, with the dominant decay mode and corresponding energy. Due to its long half-life and high fission yield, ^{137}Cs is of particular importance for offsite release; ^{134}Cs is also an important MST contributor for short-term exposures, but is generally not considered to be a long term problem from the viewpoint of land contamination [51]. Both cesium and rubidium are highly reactive, like the fellow alkali metal sodium.

Table 4-3: Dominant Isotopes of Cesium

Element	Isotope	Half-Life	Decay Mode	Energy (keV)
Cs	133	Stable	-	-
	134	2.1 yr	β^-	2058.9
	135	2.3×10^6 yr	β^-	268.90
	136	13.2 d	β^-	2548.2
	137	30.2 yr	β^-	1175.6

As described in the previous section, some cesium will form CsI within the fuel pin. However, since there is more cesium than iodine in the pin, not all cesium will form this bond and instead is likely to remain as elemental cesium and be retained by the uranium [59]. Cesium that does escape the fuel matrix will collect in the sodium bond of the fuel pin, where it will decompose from CsI to form elemental cesium. Cesium is very miscible[20] with the alkali metal of sodium. Due to this property, if the pin cladding is breached, the retention of cesium within the primary sodium is likely to be very high [52]. There has been some evidence that within sodium, cesium will collect at somewhat greater concentration near a gas-liquid boundary [51] [52].

Both elemental cesium and rubidium have a relatively high vapor pressure when compared to sodium or NaI [64] [65] and therefore are more likely to vaporize from the sodium pool. Figure 4–11 shows the equilibrium vaporization curves for both cesium and rubidium, with experimental

---

[20] The ability to form a homogeneous solution.

results for cesium. As can be seen, the vaporization of both elements is greater than that of sodium.
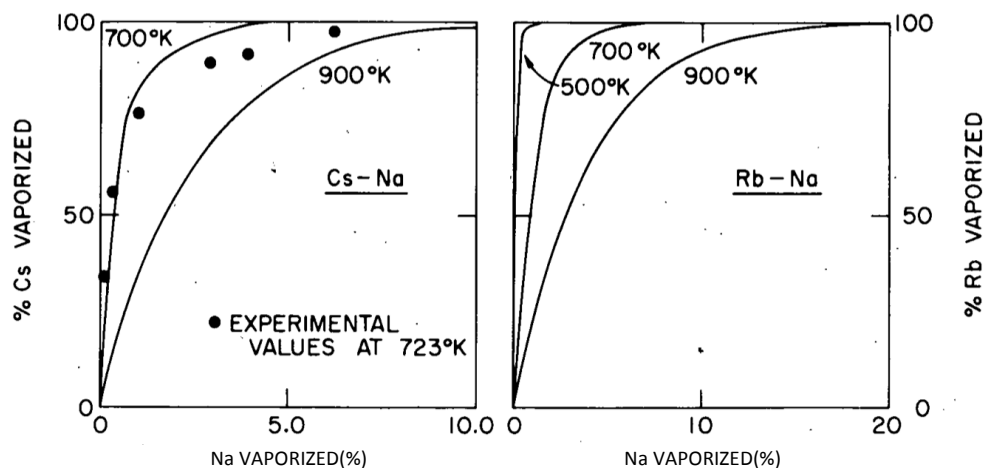
Figure 4–11: Vaporization of Cesium and Rubidium [60] [66]

Experimentation has shown a very small release fraction[21] of cesium from a hot sodium pool, on the order of 10^{-4} - 10^{-5} [67]. More recent analysis and experimentation on the vaporization of cesium has shown similar results [63] [68] [69], where the retention factor[21] for cesium in a sodium pool increases as the temperature of the liquid surface increases, as shown in Figure 4–12. This phenomenon is a result of an increase in the vaporization of sodium with higher temperature, which means the ratio of vaporized cesium to sodium decreases [69]. Even though cesium is more volatile than sodium, at expected accident sodium pool temperatures, very little vaporization of either element is likely to occur. The data appear to show cesium vaporization fractions of about one to two orders of magnitude greater than sodium vaporization, where sodium vaporization fractions are usually on the order of 10^{-4} – 10^{-6}.
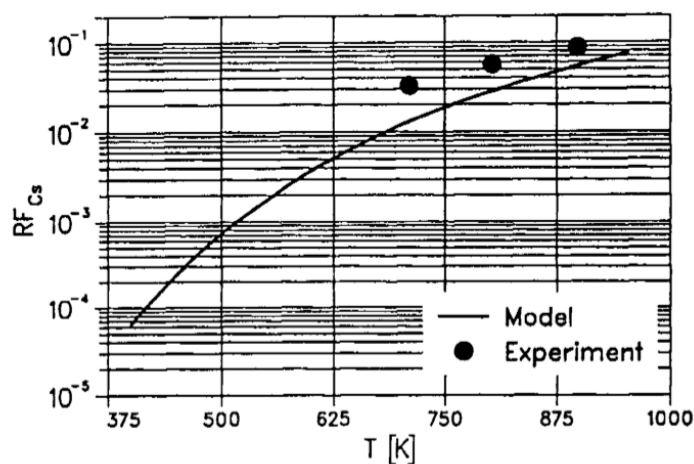
Figure 4–12: Retention Factor for Cesium in a Sodium Pool [69]

---

[21] *Release Fraction* – The ratio of the quantity of the element/compound found in the gas phase to the original quantity in the liquid phase. Perfect retention would be a release fraction of zero.

Retention Factor – The ratio of the concentration of the element/compound found in the liquid phase to the concentration in the gas phase. Perfect retention would essentially be infinity.

As will be described in Section 4.3, in past metal fuel SFR accidents, no cesium has been found in the cover gas region, despite the release of bond sodium containing cesium from numerous pins. It is not expected that significant adsorption of cesium will occur in the primary system due to its high solubility in sodium [70]. If cesium were to vaporize to the cover gas region, past experimentation has shown rapid condensation on colder surfaces [67].

4.2.4 Tellurium Group

Tellurium Group Summary
While data on tellurium and antimony behavior within the metal fuel matrix is sparse, past experiments and accidents have shown very high retention in the fuel, even with significant fuel melting. Both elements are highly soluble in sodium, with high rates of adsorption expected. The low vapor pressure and high solubility of both elements makes significant vaporization into the cover gas region unlikely.

Tellurium (Te) and antimony (Sb), which have very similar chemical properties, are the usual elements of interest from the tellurium group. In oxide fuels, these elements migrate to the cold part of the fuel matrix (the cladding interface), where high release rates (>50%) are possible when oxide fuel melts. This is not the case for metal fuel. While little information is available on the behavior of tellurium and antimony in metal fuels, past accident data along with solubility information can provide insight into their behavior during accidents.

Of all the tellurium isotopes, ^{132}Te and ^{129m}Te are of most importance. This is especially true for ^{132}Te, which decays to ^{132}I, a strong gamma emitter [41]. In past experiments, melting U-Fs alloy in a crucible for three hours at 1400°C, which is well above the melting point for metal fuel, 98.1% of the tellurium was retained in the fuel [71]. This high retention is in agreement with past SFR metal fuel accidents, where no tellurium has been found released from the fuel, even with significant melting (described in Section 4.3).

High solubility of both tellurium and antimony in sodium has been seen in past work [58]. Tellurium will form $NaTe_2$, while antimony will form $NaSb_2$ in the sodium [58]. High rates of tellurium plateout (adsorption) have been observed in past sodium reactors [58], as it has reacted with stainless steel surfaces, even at low temperatures (such as those seen in the primary sodium purification cold trap).

Tellurium has a low vapor pressure, and is expected to vaporize at a rate far below sodium or even NaI. As seen by the equilibrium vaporization curve in Figure 4–13, even with sodium vaporization nearing 100%, tellurium vaporization is expected to be well below 20% at usual sodium pool accident temperatures. Due to its low vapor pressure, even if some tellurium were to vaporize, it would be expected to quickly condense on colder surfaces. The vaporization of antimony is considered negligible, since it will be orders of magnitude less than tellurium [60].
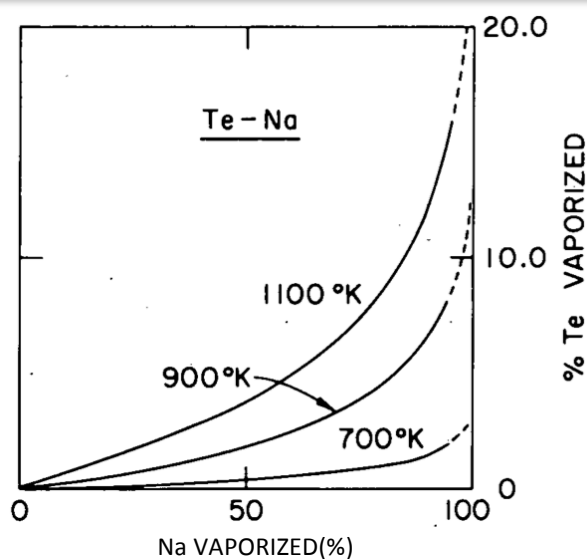
Figure 4–13: Vaporization of Tellurium [60]

4.2.5 Alkaline Earths

Alkaline Earths Summary
While there is little experimental information on the release of strontium and barium from metal fuel, both have been found in the primary sodium following past accidents. Once in the sodium, extensive adsorption is likely to occur, which will significantly reduce their concentrations in the primary sodium. Both elements have very low vapor pressures and appreciable vaporization from the sodium pool is not expected to occur.

Strontium (Sr) and barium (Ba), like the tellurium group, have noticeable differences in their behavior between oxide and metal fuel. In oxide fuel, both strontium and barium will form oxides that, when released, are not soluble in sodium [54]. However, the elemental states of both strontium and barium, which are more likely to occur in metal fuel due to the lack of oxygen, are soluble in sodium [54]. Both strontium and barium have been found in the primary sodium following metal fuel SFR accidents (described in Section 4.3).

Experimental data on the release of strontium and barium from metal fuel is sparse but in past vented metal fuel concepts (which can be taken to represent fuel performance with cladding breach), strontium and barium have not been found released from the fuel pins during normal operation [51]. Past metal fuel SFR accidents, described in Section 4.3, have found strontium and barium-lanthanum[22] in the primary sodium with plateout (adsorption) on primary system structure.

Even if strontium and barium are released to the sodium in their elemental forms, in low concentrations, both may react with oxygen impurities in the sodium and form oxides [51]. However, with larger releases, both strontium and barium will likely stay in their elemental form in sodium. Extensive plating and adsorption of strontium has been observed in experimentation [56] and in SFR accidents. The vapor pressures of strontium and barium are very low, which

---

22 ^{140}La is produced by the decay of ^{140}Ba and is found with ^{140}Ba.

should result in very little vaporization from the sodium pool, as seen in the equilibrium vaporization curves in Figure 4–14. Experimentation has seen vaporization values even lower than theoretically predicted (likely due to the formation of oxides with sodium impurities [51]).
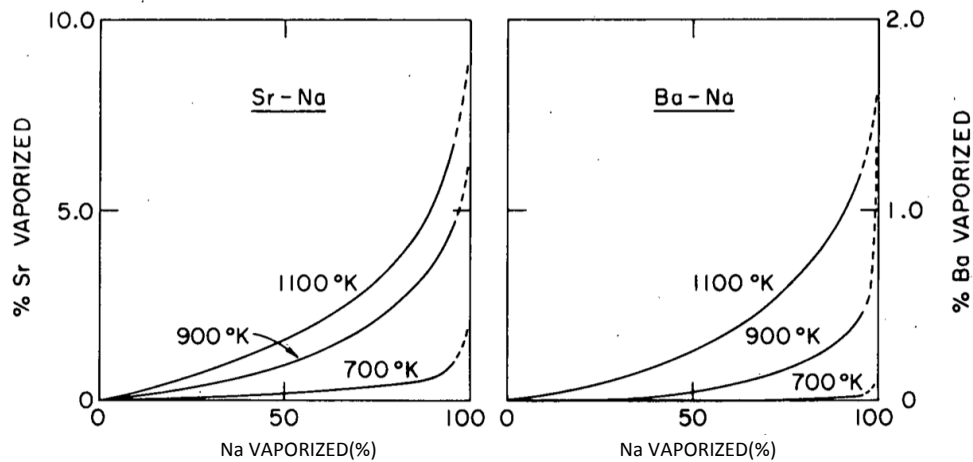
Figure 4–14: Strontium and Barium Vaporization [60]

4.2.6 Noble Metals

Noble Metals Summary
There is expected to be very little release of the noble metals from the fuel matrix, even with significant fuel melting (as observed in past experimentation). The noble metals have low solubility in sodium, and very high adsorption rates are likely, as is evident from experimentation and past reactor operation.

Information on the behavior of the noble metals (ruthenium – Ru, technetium- Tc, molybdenum – Mo, palladium – Pd, rhodium – Rh, platinum – Pt, and silver – Ag) in sodium is less refined than some of the previous radionuclide groups. However, some general observations can be made, in conjunction with limited experimental evidence.

First, the melting points of the noble metals are much higher than the expected fuel temperatures, even during most accidents. Table 4-4 summarizes the melting points for the noble metals of interest. Due to the high thermal conductivity of metal fuel and the sodium coolant, it is unlikely that fuel temperatures will reach the melting point of the noble metals (with a possible exception for silver). However, these melting points may be lower when the elements are mixed with uranium, as is the case for ruthenium [72]. Even so, experimentation has not observed significant releases, as melting of U-Fs alloys in a crucible to temperatures of 1200-1400°C, which is well above the melting point of metal fuel, for five hours showed almost complete retention of Ru, Mo, Rh, and Pd within the fuel [71][23].

---

[23] Technetium, examined in a separate experiment, was expected to act in a similar way by the authors of [70].

Table 4-4: Noble Metals Melting Point

Element	Melting Point (°C)
Ru	2334
Tc	2204
Mo	2623
Pd	1555
Rh	1963
Pt	1768
Ag	961

The noble metals are not very soluble in sodium, with the exception of silver [58], and adsorption is likely to occur. Of all the noble metals, only ruthenium has been found in the primary system during past metal fuel SFR accidents (described in Section 4.3). However, very high adsorption rates for ruthenium have been observed in experimentation (>90%) [52] and past reactor operation [58].

4.2.7 Rare Earths

> **Rare Earths Summary**
> *Cerium and yttrium are the rare earths of interest to MST development. At elevated fuel temperatures, the rare earths will combine with uranium and plutonium to form a eutectic mixture with the cladding. When in solution with uranium alone, the melting point of both elements increases at low concentration. Little data exists regarding the release fractions of cerium and yttrium from the fuel, but both are not soluble in sodium, and very high retention and adsorption is likely.*

Historically, cerium (Ce) and yttrium (Y) are the only rare earths of interest for the source term. Lanthanum is also present, but tends to accompany its parent element barium. In oxide fuels, the rare earths form soluble oxides in the fuel matrix [65]. In metal fuel, at high temperatures the rare earths, along with uranium and plutonium, will form a eutectic with the cladding [36].

While past experiments have investigated the release of rare earths from U-Fs alloy melt in a crucible [73], the presence of oxygen in the crucible caused the formation of rare earth oxides and is not indicative of metal fuel behavior during an accident. Cerium has a relatively low melting point (795 °C) but it increases when in solution with uranium [74]. The same is true for yttrium, which has a melting point of 1522 °C, when at very low concentrations in uranium [75].

The rare earths are not soluble in sodium in their elemental forms [54]. Experiments have shown high retention and adsorption (>80%) of cerium in sodium [51]. The elemental forms of the rare earths also have low vapor pressures [76], which would likely limit any possible vaporization from the sodium pool. In past SFR metal fuel accidents, cerium has been found in the primary system. However, very high retention rates have been observed due to adsorption on primary system structure (described in Section 4.3).

4.2.8 Actinides

Actinides Summary
Unlike oxide fuels, metal fuels are compatible with sodium (which is why sodium is also used as bonding material within the fuel pin). During a cladding breach and even fuel melting, there is little interaction between the fuel and sodium. In turn, very little uranium or plutonium are likely to be released into the primary sodium. While experimental evidence is sparse, both elements likely have very low solubility in sodium.

One of the main motivations behind the use of metallic fuel is its compatibility with the sodium coolant, as sodium is even used as the bond material within the pin. The compatibility between metal fuel and sodium also permits the use of vented fuel designs. This is in contrast to oxide fuel (UO_2, PuO_2), which due to the presence of oxygen, can react with sodium [58]. As will be described in Section 4.3, multiple past metal-fuel SFR accidents and breached cladding experiments have shown little metal-fuel/sodium interaction.

Data on the solubility of elemental uranium and plutonium in sodium are sparse, as most experimentation has focused on the release of UO_2 and PuO_2 from oxide fuels, which would occur more readily during cladding failure in oxide fuels. However, even elemental uranium and plutonium could form oxides in the sodium if oxygen impurities exist above 1/100 of a ppm in the sodium [77]. What data that does exist appear to show that the solubility of uranium and plutonium in sodium is very low [77] [78].

Experimentation investigating the solubility of plutonium in sodium over a prolonged period [79] shows that when using metal fuel, it may be possible for some very small fraction of the plutonium in the fuel matrix to dissolve in the sodium bond during normal operation. If this were to occur, then any dissolved plutonium would be released to the primary sodium during a cladding breach.

As will be described in Section 4.3, in past metal fuel SFR accidents, no uranium or plutonium has been found in the primary sodium or on primary system structure, even when substantial fuel melting has occurred.

4.3 Past Accidents, Experiments, and Analyses

In the U.S., there have been three sodium reactor accidents and two reactor experiments that provide specific insight into the behavior of radionuclide released from the core of a metal-fuel sodium reactor. This section briefly reviews each accident and experiment and describes the key observations as they pertain to source term development.

Sodium Reactor Experiment (SRE) Incident	Fermi 1 Incident	EBR-II Failed Fuel Capsule	EBR-II Run-Beyond-Cladding-Breach Tests	TREAT TOP Tests
1959	1966	1967-1968	1985	1984-1987
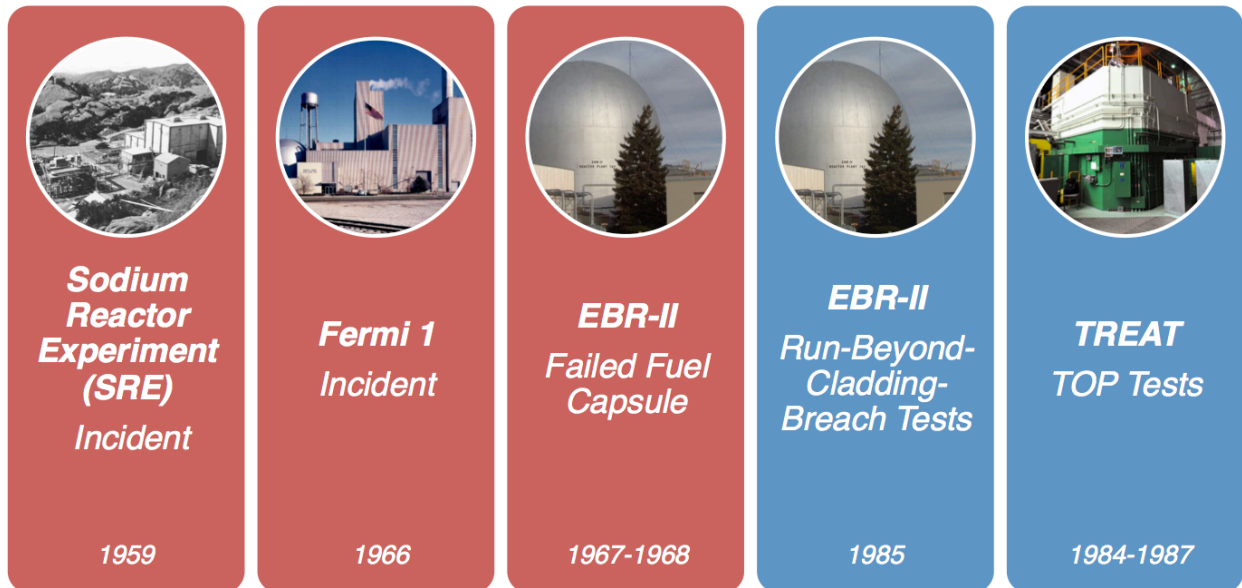
Also, the final part of this section reviews an attempted partial mechanistic source term analysis for the ALMR design. It represents a detailed chemical thermodynamic assessment of the behavior of radionuclides in the primary sodium and cover gas, and provides insight into the approximate release fractions of the radionuclide groups.

4.3.1 Sodium Reactor Experiment (SRE) Incident

> **SRE Incident Summary**
> *The SRE experienced fuel failure in 13 of its 43 metal fuel elements due to a flow blockage. Despite this release, only the noble gases were found in the cover gas, with high retention of other radionuclides in the fuel or primary sodium. However, the presence of carbonaceous material in the primary system, which is not typical for an SFR, likely aided in radionuclide retention.*

The Sodium Reactor Experiment (SRE) was a 20 MW$_{th}$ graphite moderated, sodium-cooled thermal reactor that began operation in 1957. The fuel, an unalloyed uranium metal, was enriched to 2.8% ^{235}U with a NaK bond and Type 304 stainless steel cladding. SRE also contained one experimental fuel element containing oxide fuel and five elements containing uranium alloys (U, Zr, Th, Mo combinations). The 43 fuel elements each had seven fuel rods, as seen in Figure 4–15. The core sat in a tank of sodium, with inlet and outlet piping creating a loop design.
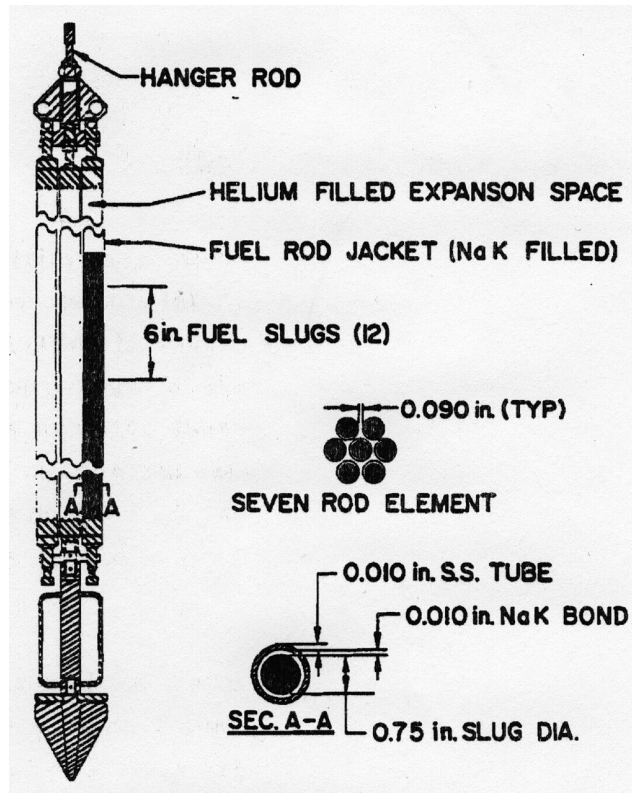
Figure 4–15: Typical SRE Fuel Element [80]

Around July 12th, 1959, a tetralin[24] leak into the primary sodium coolant reacted to form a particulate that restricted flow in the core and caused overheating and damage to 13 of the reactor's 43 fuel elements. It is unknown when exactly the fuel damage first occurred, but the damage likely began around July 12-13, shortly after a restart, with the majority of failures occurring between July 21-23 [81]. Despite abnormal temperature and radioactivity readings, the reactor was not shutdown until July 23.

While cladding failures occurred in 13 fuel elements (11 unalloyed uranium fuel elements, and two uranium alloy fuel elements) [80], no significant uranium melting was found beyond that which occurred as part of the eutectic formation with iron [81]. Figure 4–16 shows the fuel damage in one of the fuel elements (channel 55) containing the experimental uranium alloy fuel.

Through an examination of the failed elements and thermocouple readings, it is thought that 11 of the fuel elements failed due to eutectic cladding penetration, while two failed due to repeated cycling through the $\alpha - \beta$ phase transformation temperature for uranium, which caused the fuel to expand until the cladding burst (as was the case for channel 55) [80]. Another possibility is that boiling of the NaK fuel bond, which has a lower boiling point than sodium, may have put additional pressure on the cladding. The low burnup of the fuel at the time of failure (less than 0.1% core average) likely limited the release of fission gas, since the gas bubbles had not yet agglomerated [53].

---

[24] Tetralin is an oil-like hydrocarbon that was used to cool the primary pump seals.
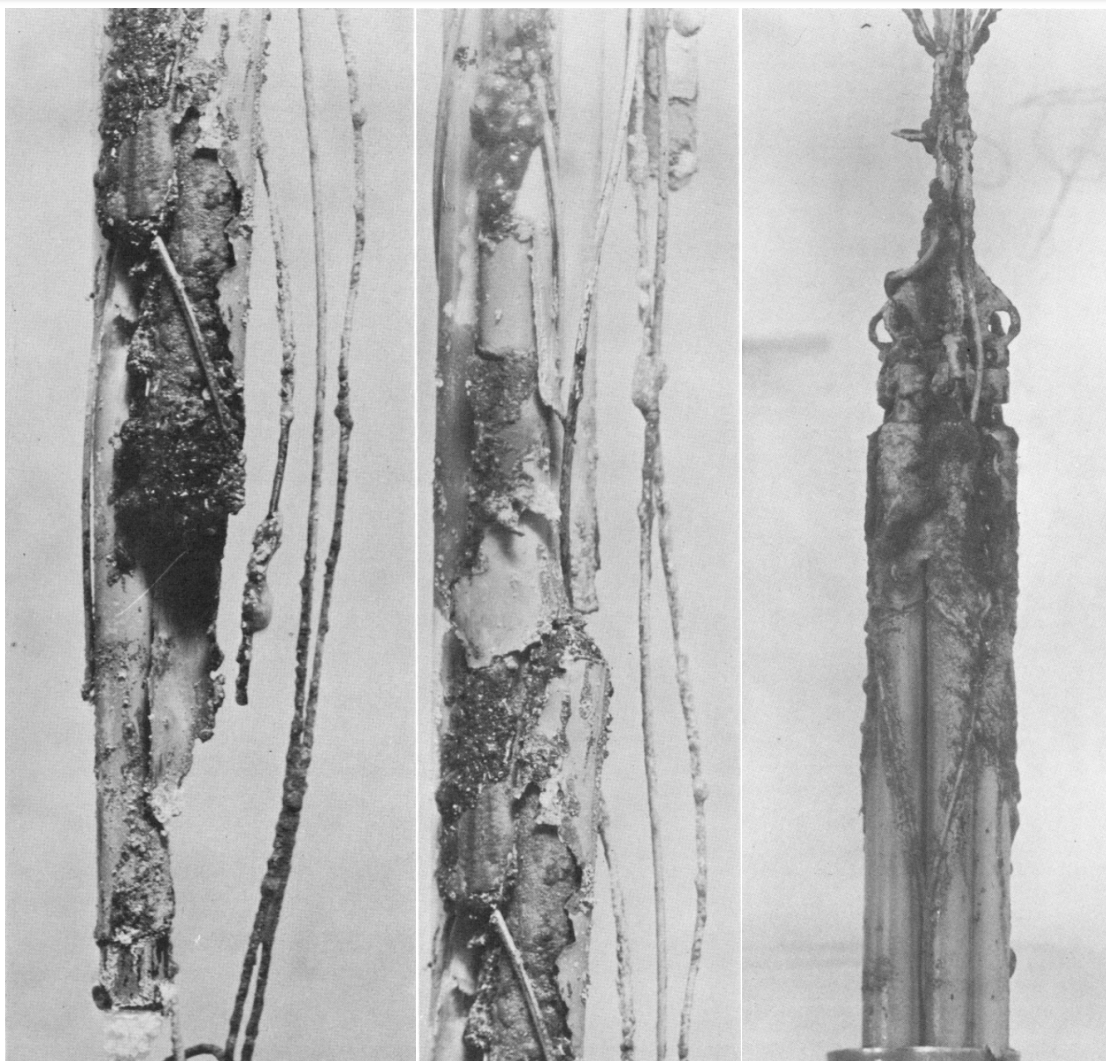
Figure 4–16: Fuel Damage in Channel 55 of SRE (Bottom, Middle, Top)[25] [80]

The release at SRE is somewhat unique due to the presence of large quantities of carbon in the reactor. It is estimated that 7 to 70 lbs of carbonaceous material may have been deposited in the primary system as a result of the tetralin leak [81]. The carbon likely acted as a filter, removing some of the radionuclides within the primary sodium, as subsequent radioactive analysis of the carbon fragments showed a much higher concentration of fission product contamination than in the primary sodium.

Only Xe and Kr were found in the cover gas after the accident, with Cs, Sr, I, Ce, Ba-La, Zr-Nb, and Ru in the primary sodium [81]. Strontium was by far the biggest contributor to activity in the primary sodium, but again, the presence of carbonaceous material may have influenced the behavior of radionuclides. Subsequent analysis of the primary system showed substantial plateout of Sr and Ce on primary piping, but not of Cs, since it is a fellow alkali metal of sodium [81].

---

[25] Channel 55 contained experimental uranium alloy fuel pins. In the pictures, the wire wrapping has detached from the fuel pins due to cladding failure.

An investigation into the SRE incident noted several key conclusions [81]; of particular interest are the following:

1) Although significant fuel melting did not occur, the release fraction of the various fission products to the primary sodium indicate some degree of volatile fission product release.
2) Only Xe and Kr isotopes were identified in the reactor cover gas system.
3) The iodine release fraction was smaller than expected, and deposition in the primary system or escape to the cover gas occurred in undetectable quantities[26].
4) The carbonaceous particulate material that resulted from the tetralin leak proved to be an effective fission product scavenger[27].

SRE was later repaired and resumed operations in late 1960, and continued to operate until 1964. Boeing, who later purchased the SRE site, was the target of a 2004 class action lawsuit claiming harm to local residents due to the 1957 accident. In particular, the plaintiffs argued that significantly more iodine had been released from the plant than what was documented in official estimates. The court case resulted in renewed interest in metal-fuel radionuclide release phenomena, and analyses prepared by expert witnesses on behalf of Boeing [53] [82] concluded that there was no significant release of iodine due to retention within the fuel and primary sodium.

4.3.2 Fermi 1 Incident

Fermi 1 Incident Summary
Fermi 1, a commercial SFR, experienced significant fuel melting in two assemblies, and minor damage in two others, due to a flow blockage. It is estimated that ~150 fuel pins melted in total. Even though major fuel melting and relocation occurred, only the noble gases were found in the cover gas region, with other radionuclides retained in the fuel or primary sodium. The reactor was later restarted with an oxide core following a cleanup and repair effort.

The Fermi 1unit, located at the current site of the Enrico Fermi Nuclear Generating Station outside of Detroit, was a 200MW$_{th}$/66MW$_e$ sodium fast breeder reactor operated by Detroit Edison. The intent of Fermi 1 was to demonstrate the commercial viability of a fast reactor power station. The reactor went online in 1963 and began generating power in mid-1966. Fermi 1 used U-10wt% Mo fuel pins with zirconium cladding that were enriched to 25.6% ^{235}U. It contained 105 core subassemblies with 140 pins per square subassembly [83]. The core sat in a sodium-filled vessel, with inlet and outlet piping as part of a loop SFR design.

On October 5th, 1966, two pieces of a Zircaloy baffle broke loose from the melt-down section liner below the core. This Zircaloy device for retaining molten core had been a late addition to the Fermi 1 design [83]. The pieces of Zircaloy were carried upward by the primary sodium flow and were lodged at the entrance of the core region. The blockage starved coolant flow to the core and

---

[26] This is likely due to the formation and retention of UI$_3$ in the fuel, which was not well understood at the time of the SRE investigation report.

[27] This is atypical of a sodium fast reactor accident.

resulted in damage to four of the 105 subassemblies. Significant fuel melting occurred in two subassemblies, where it is theorized that flow was reduced to approximately 3% of nominal [83]. The other damaged subassemblies likely had flow reduced to 10% and 30% of nominal [84]. The entire incident evolved in one hour during a rise in reactor power (to 31 MW$_{th}$) as part of a plant test [83]. All four subassemblies had been in the reactor since the original loading program[28]. In the two subassemblies that experienced melting, M-127 and M-098, molten fuel slumped to the bottom of the assembly as melting progressed [83].

Figure 4–17, shows the damage to subassembly M-127 (which had been cut axially for analysis), and subassembly M-098, which was noticeably more damaged, with significant breaches to the subassembly can. Subsequent analyses determined that once melting in M-127 and M-098 penetrated the subassembly can walls, stagnant sodium present between the subassemblies provided convective, two-phase cooling to the subassembly interior and prevented further damage [84]. It should be noted that no propagation of fuel melting to neighboring assemblies was seen [83].
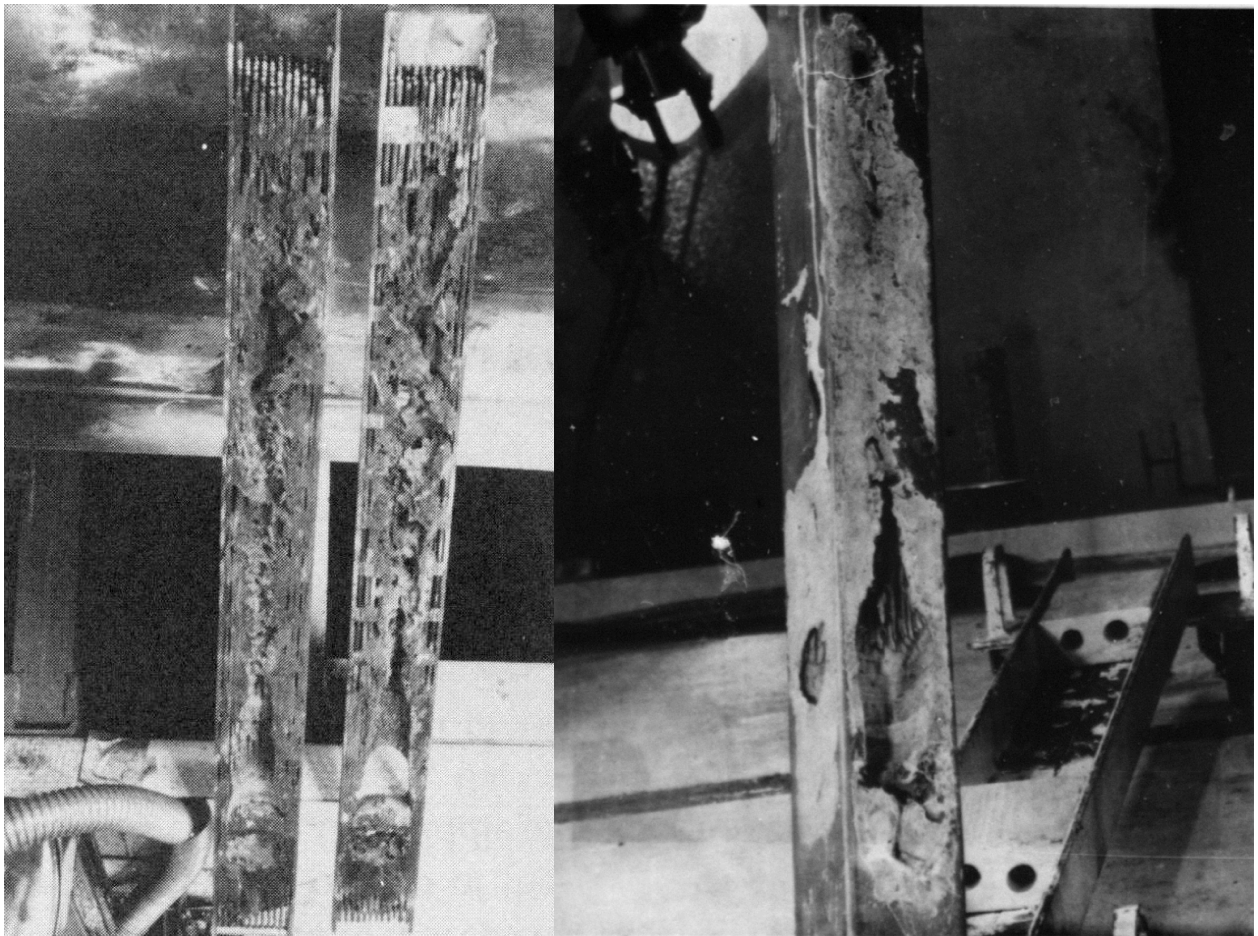
Figure 4–17: Fermi 1 - Subassembly M-127 (Left), Subassembly M-098 Damaged Can Faces (Right) [83]

---

28 About 25,000 hours in core, but most of the time was dedicated to reactor testing. The reactor had only begun power generation two months prior to the accident [82]. The estimated burnup was < 0.05 a/o [83].

As a result of the accident, Xe and Kr were both present in the reactor cover gas. Within the primary sodium, Cs, I, Sr, and Ba-La were detected. A small amount of Xe and Kr was released into the containment building through ordinarily insignificant leaks in the primary cover gas system, but radioactivity levels were not dangerously high, and no significant plant personnel exposure occurred. The containment building isolation system automatically activated and prevented any significant release through the waste stack [85].

After the accident, but before the damaged subassemblies were removed from the core, estimations of damage were calculated based on the quantity of radionuclides present in the cover gas and primary sodium. The results of this analysis can be seen in Table 4-5. Since the release fractions of radionuclides from the fuel were unknown, ranges of estimations were provided. A reason for the high ^{89}Sr readings was not provided.

Table 4-5: Fermi 1 – Fuel Failure Analysis for Radioactivity [83]

Region	Isotope	Activity in System (Ci)	Activity in Average Pin (Ci)	Number of Fuel Pins Melted per Assumed Release Fraction		
				10%	50%	100%
Cover Gas	Kr-85	1.43	0.02		142	71
	Xe-133	6.4	0.118		109	54
					240**	170*
Primary Na	Cs-137	2.75	0.17	162	32.4	16.2
	Sr-89	1570	5	3140	628	314
	I-131	5.04	0.33	153	30.6	15.3

* 100% release of ^{133}Xe and 0% release of ^{133}I (which would lead to additional ^{133}Xe)
** 50% Release of ^{133}Xe, 10% release of ^{133}I

Using this information, the approximate release fractions of radionuclides can be determined, as presented in Table 4-6 assuming melting of an equivalent to one (140 pins) or two (280 pins) subassemblies. Later analysis of the removed subassemblies estimated fuel melting at approximately a little over one subassembly worth of fuel pins [83].

Table 4-6: Fermi 1 Estimated Release Fractions Based on Findings in Ref [83]

	Estimated Release Fraction to Primary Sodium	
Isotope	140 Pin Melt	280 Pin Melt
Kr-85	~50%	~10%
Xe-133	~100%*	<50%**
Cs-137	~10%	~5%
Sr-89	Inconclusive	Inconclusive
I-131	~10%	~5%

* Assuming 0% ^{133}I release
** Assuming 10% ^{133}I release

Further analysis conducted by the operators at Fermi 1 found high plateout losses of Sr and Ba-La in the primary sodium, as shown in Table 4-7. As expected, plateout of the alkali metal Cs was comparably low. These findings are similar to what was observed at SRE. However, no significant plateout in the heat exchangers was found [56].

Table 4-7: Fermi 1 – Estimated Plateout Losses [64]

Isotope	Estimated Plateout Loss
Sr-89 & Sr-90	93%
Cs-137	8%
Ba-La-140	75-84%

A significant cleanup and repair operation was initiated after the Fermi 1 accident, and the metallic U-Mo fuel was replaced with an oxide core. Fermi 1 was restarted in 1970 and continued to operate until 1972.

4.3.3 EBR-II Fuel Failure Incident

EBR-II Fuel Failure Incident Summary
A metal fuel element, within an experiment capsule, endured repeated melting over the course of five months due to a defect in the fuel that led to insufficient cooling. Significant melting and fuel relocation occurred in the fuel element as EBR-II cycled through power operations. Over the course of the incident, only the noble gases were found in the cover gas region, with other radionuclides retained in the fuel and primary sodium.

As described in Section 2, EBR-II was a 62.5 MW_{th} SFR built and operated by Argonne National Laboratory at the current location of INL. The reactor operated from 1963 to 1994. From November 23, 1967, to March 1968, a series of radionuclide releases occurred due to the melting of an U-Pu-Zr fuel element within an experiment capsule.

Capsule BC02, pictured in Figure 4–18, contained an experimental ternary metallic alloy fuel element composed of U-75% Pu-15% Zr-10%. Due to two flaws that likely occurred in the manufacturing process, during irradiation, sodium was lost from the capsule bond. Adequate cooling could not be maintained due to the inability to transfer heat from the fuel [86]. On November 23, 1967, during the first reactor startup with BC02 in core, a spike in the ^{135}Xe activity in the cover gas was detected, as seen in Figure 4–19. However, the reading for ^{133}Xe did not increase significantly. This indicated a fresh fuel failure, since ^{133}Xe is a longer-lived radionuclide compared to the short-lived ^{135}Xe.

Operations continued, and as the reactor was repeatedly cycled through power operations and shutdown, which was common for EBR-II, the longer-lived ^{133}Xe became present by December 1967. By April of 1968, increases in the ^{131}I level in the primary sodium started to occur. A search began to find the failed fuel element. Through process of elimination, the experimental subassembly containing BC02 was removed in May 1968 [87].

Subsequent evaluation and radiograph of BC02, seen in Figure 4–20, indicated that gross fuel melting and rearrangement had occurred in the upper and lower regions of the fuel pin. The molten fuel flowed, or slumped, outward until it contacted the colder capsule walls [87]. It is likely that multiple melting events occurred over the ~7 month timeframe that BC02 was in the core. The increase in ^{131}I levels in the primary sodium, seen at the start of 1968, was likely the result of the release of any remaining bond sodium from the fuel element, which had become chemically fixed with small amounts of iodine from the fuel during melting [87].
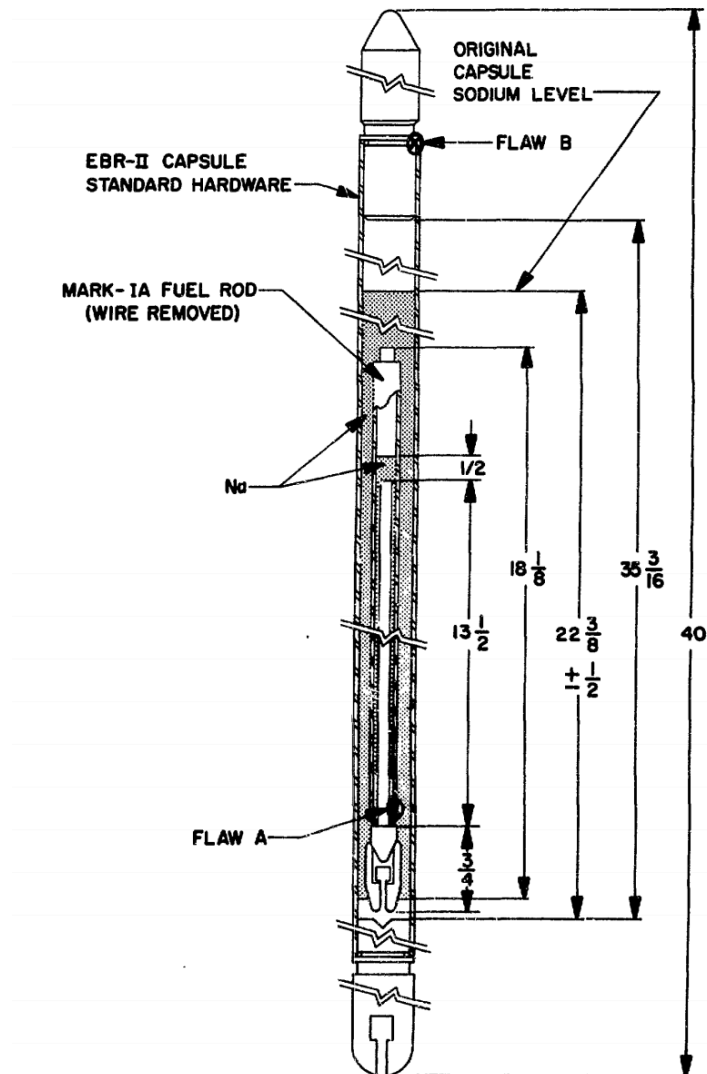
Figure 4–18: Diagram of Capsule BC02 (with Two Flaw Locations Highlighted) [87]
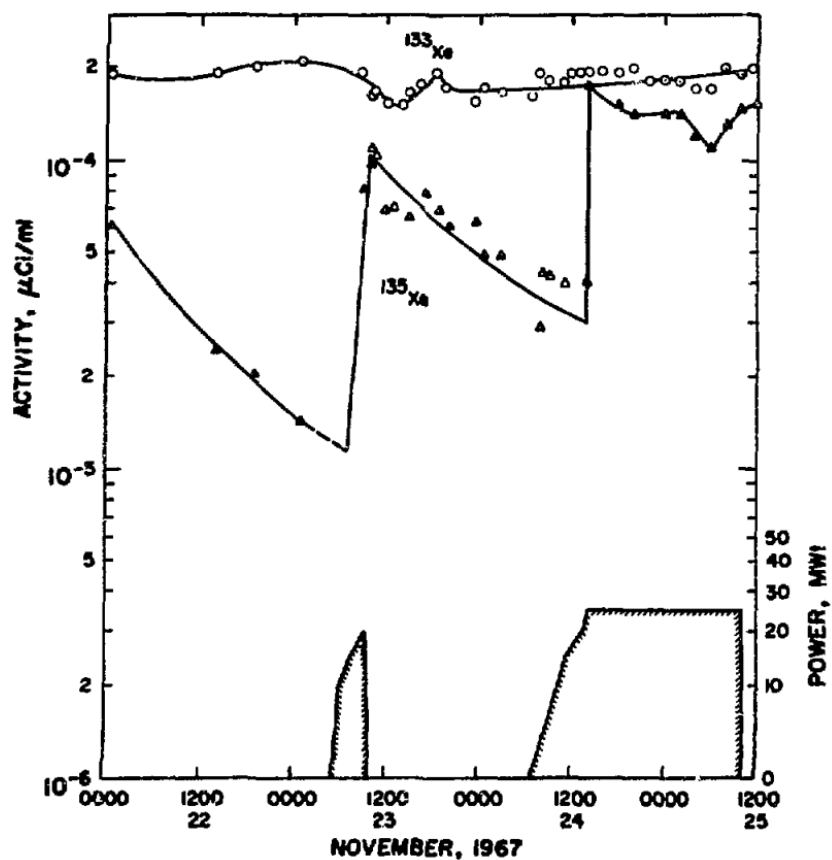
Figure 4–19: EBR-II Cover Gas Samples – November 1967 [87]

Neutron Radiograph of Upper
Section of Capsule BC02

Neutron Radiograph of Mid-
section of Capsule BC02

Neutron Radiograph of Lower
Section of Capsule BC02
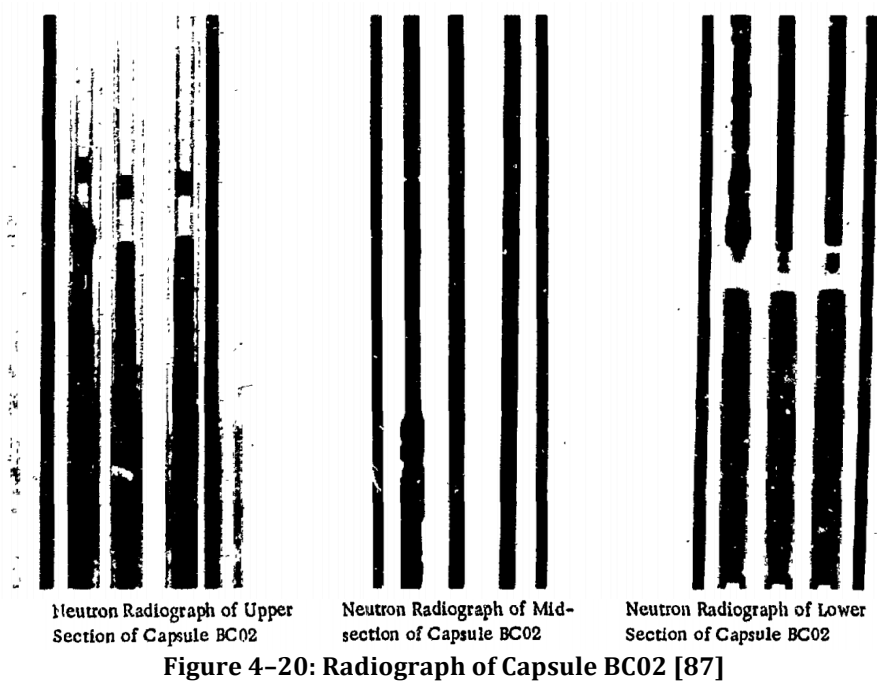
Figure 4–20: Radiograph of Capsule BC02 [87]

4.3.4 EBR-II Run-Beyond-Cladding-Breach Tests

Run-Beyond-Cladding-Breach Tests Summary
A series of experiments were conducted at EBR-II to gauge the effects of a breach in the cladding of a fuel pin. The cladding of pre-irradiated fuel pins was purposely degraded in order to induce cladding failure when the pin was reintroduced to the core. Once clad failure occurred, the pins were left in the reactor for several months. Fission gases and bond sodium were released when the cladding failed. However, the results showed excellent compatibility between the metal fuel and sodium coolant, with no detectable releases of fuel material to the primary sodium.

A series of experiments, referred to as the Run-Beyond-Cladding-Breach (RBCB) tests, were performed at EBR-II with metal fuel elements with intentionally weakened cladding. The goal of these experiments was to analyze the behavior of metal fuel after a cladding breach occurs and to demonstrate the compatibility between metal fuel and sodium. Tests were performed with U-Fs, U-Zr, and U-Pu-Zr fuel elements with various types of cladding [35].

The RBCB tests were performed by grinding down an area of the cladding surface of a pre-irradiated pin until only 30-40 μm of the cladding remained [35]. The pin was then reinserted into the reactor, with cladding failure occurring shortly after reinsertion as pin pressure increased. The reactor would then continue to operate with the failed pin in core. A summary of the RBCB test results can be seen in Table 4-8.

Table 4-8: Summary of RBCB Tests at EBR-II [35]

Experiment ID Number	XY-21A	XY-24	XY-27	X-482	X482A	X482B
Composition, wt%	U-5Fs	U-19Pu-10Zr	U-19Pu-10Zr	U-19Pu-10Zr	U-10Zr	U-19Pu-10Zr
Cladding material	316SS	316SS	316SS	D9	D9	HT9
Final burnup, at%	~9.3	~7.5	~6.0	14.4	13.5	13.5
Pin diameter, mm	4.4	4.4	4.4	5.8	5.8	5.8
No. of days breached	54	233	131	168	100	150
DN signal, cps (note a)	~30-40	(note b)	(note b)	~600	~700	(note c)
Weight loss, g (note d)	2.0	2.7	2.5	4.0	3.6	3.9
Peak cladding temp(°C).	550	550	550	600	600	600

Notes: (a) Counts above background.
(b) Unavailable due to instrument malfunction.
(c) None detected, breached at startup.
(d) Expulsion of bond sodium fission gas and cesium accounts for the weight loss; fuel loss was negligible.

With the breach in cladding, fission gases and bond sodium were released from the pin. Cesium was also expelled into the primary coolant, as it had dissolved in the bond sodium. The expulsion of fission gas produced a virtually complete depressurization of the pin [35], resulting in no breach propagation (even after many days of continued operation). No fuel was extruded from the breach during any of the RBCB tests [88]. As shown in Figure 4–21, due to the metal fuel compatibility with sodium, no significant reaction occurs at the breach location, in contrast to oxide fuel, where the reaction with sodium exacerbates the cladding breach.
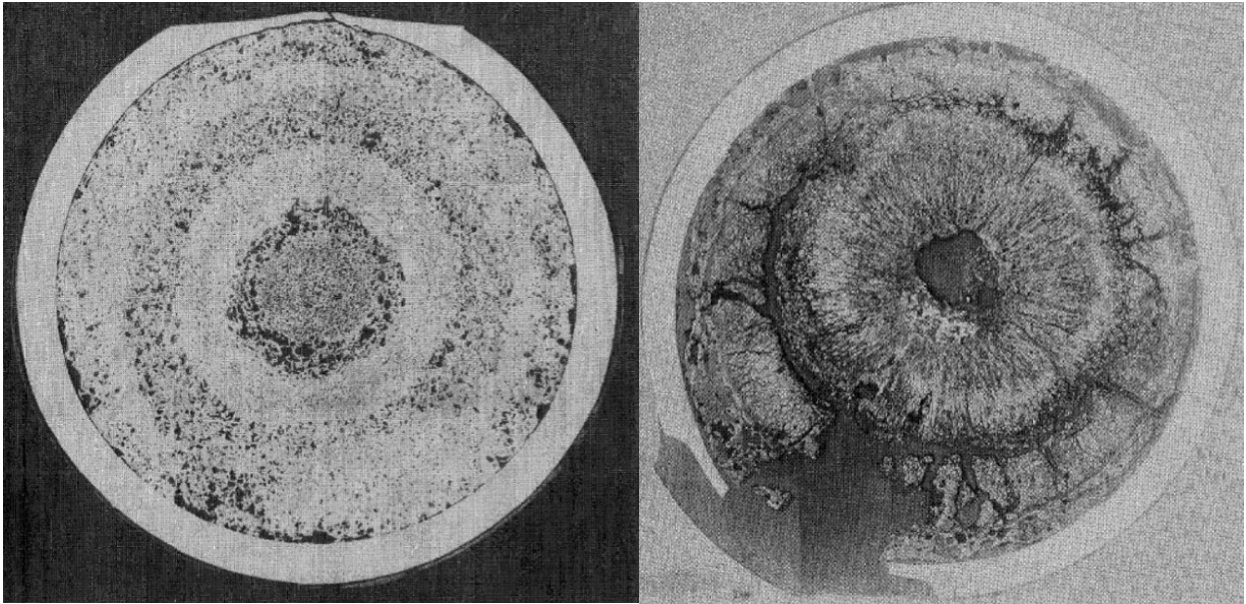
Figure 4–21: Comparison of Breach in Cladding for Metal Fuel (Left) and Oxide Fuel (Right) [27]

4.3.5 TREAT Tests

TREAT Tests Summary
Metal fuel pins were subjected to severe transient overpowers in the TREAT test reactor. Of the 15 metal-fuel pins tested, all experienced internal fuel melting, but only five overheated to the point of cladding breach (which occurred at ~4 times nominal power). Cladding breach always occurred at the top of the pin (the hottest location of cladding), and was accompanied by fuel sweep-out and freezing. In the cases with no cladding failure, there was no radionuclide release from the fuel pins despite internal fuel melting.

In the mid-1980s, a series of safety tests was performed on metal fuel at the Transient Reactor Test (TREAT) facility at the current location of Idaho National Laboratory. These tests investigated the failure of metal fuel pins during transient overpower scenarios. The objective was to study the behavior of fuel and cladding near the cladding failure threshold for a range of burnup values and fuel/cladding combinations [35].

Beginning in 1985, tests M2 through M7 subjected metal fuel pins to a transient overpower with an eight second period. The fuel pins were placed within a tube that allowed sodium coolant flow throughout the test. Internal fuel melting occurred in all 15 of the fuel pins subjected to testing, however, only five pins were overheated to the point of cladding breach, which occurred in the range of slightly over four times nominal power. The breaches likely occurred due to the onset of rapid eutectic penetration of the cladding as fuel temperatures increased in conjunction with increasing internal pin pressure due to the expansion of fission gases within the pin or boiling of the sodium in the bond region [35].

The major findings of the tests indicated that the high thermal conductivity of the metal fuel assured that peak cladding temperatures occurred near the top of the fuel column, where the coolant is the hottest. For the TREAT tests, this meant that cladding breach also occurred near the

top of the fuel pin. It should be noted that this may not be the case for other metal-fuel pin designs with significantly different power profiles [36].

Of the five pins that experienced cladding breach, about half of the fuel inventory was ejected from the fuel pin as molten material through the (small) breach at the top of the pin. The fuel material was then swept out of the fuel region and deposited on surfaces of the sodium test loop upstream of the active region of the test fuel [89]. No significant flow blockages were observed from the ejected fuel.

In terms of radionuclide transport, no radionuclides were found for those tests where the cladding of all pins remained intact, despite fuel melting within the pin. For the tests where cladding breach occurred, ^{89}Rb, ^{138}Cs, and ^{138}Xe were detected in the loop, likely as a result of the following chain [89]:

1) Fission products ^{89}Br and ^{138}I, which are soluble in sodium, were released from the fuel and transported efficiently in the coolant.
2) ^{89}Br and ^{138}I decayed into noble gases, ^{89}Kr and ^{138}Xe, which escaped into the gas plenum.
3) ^{89}Kr and ^{138}Xe decayed into ^{89}Rb and ^{138}Cs, which settled out on the test loop wall.

4.3.6 ALMR Source Term Analysis

ALMR Source Term Analysis Summary
A partially mechanistic source term assessment was performed as part of the ALMR project. Thermodynamic equilibrium calculations, which are not without limitations, were performed to assess the migration of radionuclides within the fuel pin and in the primary sodium and cover gas. The results appear to closely align with past experimentation and accidents, with complete transport of the noble gases, but very high retention rates of essentially all other radionuclides, even without accounting for adsorption and mechanical deposition within the primary system.

As part of the ALMR project in the mid-1990s, a partial mechanistic source term analysis was conducted by ECN (Energy research Center of the Netherlands) [55]. While the analysis postulated an accident and releases from the fuel, chemical thermodynamic calculations were performed regarding the behavior of radionuclides in the sodium and cover gas.

The analysis considered two scenarios:

- **Case 1** – Examined the release of radionuclides from the fuel into the fission gas plenum of the fuel pin for a single pin at normal operating conditions (575°C).
- **Case 2** – Examined the release of the entire inventory of the core to the primary sodium at the boiling point of sodium (883°C) and the resulting fraction released to the cover gas.

Thermodynamic equilibrium calculations were performed using available Gibbs free energy data (more information on thermodynamic calculations can be found in Appendix A), assuming homogeneous mixing with no temperature gradients. This strategy is not without its faults, as

equilibrium and perfect mixing may overestimate the retention of radionuclides, since homogeneity may not occur over short time frames, especially in releases to the primary sodium. Also, thermodynamic data were not available for every radionuclide (the study highlighted a lack of lanthanide and actinide data [55]). If data were lacking, a bounding assumption of no mixing could be taken, but this can significantly underestimate retention.

A second possible fault is that equilibrium calculations assume all elements will form the most thermodynamically preferential compounds. As discussed in Section 4.2.2, even though the formation of CsI is thermodynamically preferred compared to the formation of UI_3, the much greater quantity of uranium in the fuel pin is likely to lead to some UI_3 formation. Similar phenomenon may be possible for other elements.

The results of Case 1, the pin fission gas plenum analysis, can be found in Table 4-9. The largest release fraction is found with the noble gases, which are expected to completely migrate to the fission gas plenum. The alkali metals of cesium and rubidium are also found in the fission gas plenum, but at release fractions that are orders of magnitude less than the noble gases. Iodine also transports to the fission gas plenum, but in compound with cesium. Europium, a lanthanide, is the only other radionuclide to have a relatively high release fraction. Europium is an element that has not received much analysis is past experimentation, but appears to migrate due to its high vapor pressure. However, as mentioned earlier, data on lanthanide behavior is sparse.

Table 4-9: ALMR Source Term Results – Case 1 [55]

Element	Inventory Mole	Plenum Gas Mole	Gram	Release Fraction	Chemical Form
He	$2.17 \cdot 10^{-3}$	$2.17 \cdot 10^{-3}$	$0.87 \cdot 10^{-2}$	1.00	He
Kr	$1.88 \cdot 10^{-3}$	$1.88 \cdot 10^{-3}$	$1.57 \cdot 10^{-1}$	1.00	Kr
Xe	$2.20 \cdot 10^{-2}$	$2.20 \cdot 10^{-2}$	2.89	1.00	Xe
O	$0.94 \cdot 10^{-6}$	$2.77 \cdot 10^{-24}$	$0.44 \cdot 10^{-22}$	$2.95 \cdot 10^{-18}$	CeO
H	$1.49 \cdot 10^{-6}$	$1.49 \cdot 10^{-6}$	$1.50 \cdot 10^{-6}$	1.00	H_2
I	$1.84 \cdot 10^{-3}$	$2.96 \cdot 10^{-8}$	$0.38 \cdot 10^{-5}$	$1.61 \cdot 10^{-5}$	CsI, Cs_2I_2
Br	$1.03 \cdot 10^{-4}$	$0.47 \cdot 10^{-10}$	$0.37 \cdot 10^{-8}$	$0.46 \cdot 10^{-6}$	$NaBr, Na_2Br_2$
Na	0.34	$1.55 \cdot 10^{-5}$	$0.36 \cdot 10^{-3}$	$0.46 \cdot 10^{-4}$	Na, Na_2
Cs	$1.85 \cdot 10^{-2}$	$1.77 \cdot 10^{-5}$	$2.36 \cdot 10^{-3}$	$0.96 \cdot 10^{-3}$	Cs,
Rb	$1.56 \cdot 10^{-3}$	$1.99 \cdot 10^{-6}$	$1.70 \cdot 10^{-4}$	$1.27 \cdot 10^{-3}$	Rb, Rb_2
Te	$0.32 \cdot 10^{-2}$	$2.21 \cdot 10^{-19}$	$2.82 \cdot 10^{-17}$	$0.70 \cdot 10^{-16}$	CsTe
Sb	$0.36 \cdot 10^{-3}$	$0.64 \cdot 10^{-9}$	$0.77 \cdot 10^{-7}$	$1.78 \cdot 10^{-6}$	Sb_4, Sb_2, Sb
Ba	$0.64 \cdot 10^{-2}$	$2.81 \cdot 10^{-11}$	$0.39 \cdot 10^{-8}$	$0.44 \cdot 10^{-8}$	Ba
Sr	$0.36 \cdot 10^{-2}$	$0.76 \cdot 10^{-9}$	$0.67 \cdot 10^{-7}$	$2.12 \cdot 10^{-7}$	Sr
Ru	$2.02 \cdot 10^{-2}$	$1.21 \cdot 10^{-35}$	$1.22 \cdot 10^{-33}$	$0.60 \cdot 10^{-33}$	Ru
Mo	$2.01 \cdot 10^{-2}$	$1.86 \cdot 10^{-36}$	$1.79 \cdot 10^{-34}$	$0.92 \cdot 10^{-34}$	Mo
La	$0.54 \cdot 10^{-2}$	$0.87 \cdot 10^{-23}$	$1.20 \cdot 10^{-21}$	$1.61 \cdot 10^{-21}$	La
Ce	$1.10 \cdot 10^{-2}$	$1.46 \cdot 10^{-23}$	$2.05 \cdot 10^{-21}$	$1.33 \cdot 10^{-21}$	Ce, CeO
Eu	$0.55 \cdot 10^{-3}$	$0.45 \cdot 10^{-8}$	$0.68 \cdot 10^{-6}$	$0.81 \cdot 10^{-5}$	Eu
Y	$1.88 \cdot 10^{-3}$	$0.69 \cdot 10^{-22}$	$0.62 \cdot 10^{-20}$	$0.37 \cdot 10^{-19}$	Y
Zr	$1.82 \cdot 10^{-2}$	$0.42 \cdot 10^{-34}$	$0.39 \cdot 10^{-32}$	$2.32 \cdot 10^{-33}$	Zr
Pu	$2.36 \cdot 10^{-1}$	$0.35 \cdot 10^{-19}$	$0.86 \cdot 10^{-17}$	$1.48 \cdot 10^{-19}$	Pu
Np	$1.03 \cdot 10^{-2}$	$2.85 \cdot 10^{-26}$	$0.67 \cdot 10^{-23}$	$2.76 \cdot 10^{-24}$	Np
U	1.12	$0.40 \cdot 10^{-28}$	$0.94 \cdot 10^{-26}$	$0.35 \cdot 10^{-28}$	U, UI_2
Am	$1.57 \cdot 10^{-2}$	$3.11 \cdot 10^{-14}$	$0.76 \cdot 10^{-11}$	$1.99 \cdot 10^{-12}$	Am
Cm	$1.16 \cdot 10^{-3}$	$2.49 \cdot 10^{-21}$	$0.62 \cdot 10^{-18}$	$2.14 \cdot 10^{-18}$	Cm

The results for Case 2 can be seen in Table 4-10. Again, it is important to note that this analysis involves the release of the complete core inventory to the primary sodium at very high sodium temperatures (which maximizes vaporization) with no credit for retention within the fuel or retention due to adsorption on primary system structure. As to be expected, the results closely align with the experimental data and past accident findings described in the previous two sections. The noble gases (Xe and Kr) are almost completely transported and released to the cover gas region. The fraction of iodine released to the cover gas, in the form of NaI and some small fraction of CsI, is very small (on the order of 10^{-5}), which is to be expected due to its low vapor pressure. The cesium and rubidium release fractions are on the order of 10^{-3}, which also aligns with past experimentation and analysis. Europium again has a relatively high release fraction due to its high vapor pressure.

Table 4-10: ALMR Source Term Results – Case 2[29] [55]

Element	Inventory Mole	Cover Gas Mole	Gram	Release Fraction	Chemical Form
He	$3.10 \cdot 10^3$	$2.95 \cdot 10^3$	$1.18 \cdot 10^4$	0.95	He
Kr	$1.24 \cdot 10^2$	$1.22 \cdot 10^2$	$1.03 \cdot 10^4$	0.99	Kr
Xe	$1.46 \cdot 10^3$	$1.46 \cdot 10^3$	$1.91 \cdot 10^5$ ·	1.00	Xe
O	$0.87 \cdot 10^2$	$2.11 \cdot 10^{-10}$	$0.34 \cdot 10^{-8}$	$2.44 \cdot 10^{-12}$	CeO
H	3.10	3.10	3.12	1.00	H_2
I	$1.22 \cdot 10^2$	$1.25 \cdot 10^{-3}$	$1.59 \cdot 10^{-1}$	$1.03 \cdot 10^{-5}$	NaI,CsI
Br	$0.66 \cdot 10^1$	$2.00 \cdot 10^{-5}$	$1.60 \cdot 10^{-3}$	$3.04 \cdot 10^{-6}$	NaBr
Na	$3.02 \cdot 10^7$	$1.59 \cdot 10^3$	$0.36 \cdot 10^5$	$0.53 \cdot 10^{-4}$	Na, Na_2
Cs	$1.24 \cdot 10^3$	0.69	$0.92 \cdot 10^2$	$0.56 \cdot 10^{-3}$	Cs, CsI, Cs_2
Rb	$1.04 \cdot 10^2$	$0.46 \cdot 10^{-1}$	$0.39 \cdot 10^1$	$0.44 \cdot 10^{-3}$	Rb, Rb_2
Te	$2.10 \cdot 10^2$	$0.46 \cdot 10^{-8}$	$0.58 \cdot 10^{-6}$	$2.18 \cdot 10^{-11}$	CsTe, Te, Te_2
Sb	$2.12 \cdot 10^1$	$2.30 \cdot 10^{-7}$	$2.80 \cdot 10^{-5}$	$1.09 \cdot 10^{-8}$	Sb_4,Sb_2,Sb
Ba	$0.43 \cdot 10^3$	$2.29 \cdot 10^{-5}$	$3.14 \cdot 10^{-3}$	$0.53 \cdot 10^{-7}$	Ba
Sr	$2.35 \cdot 10^2$	$0.34 \cdot 10^{-3}$	$2.95 \cdot 10^{-2}$	$1.43 \cdot 10^{-6}$	Sr
Ru	$1.34 \cdot 10^3$	$1.02 \cdot 10^{-18}$	$1.03 \cdot 10^{-16}$	$0.76 \cdot 10^{-21}$	Ru
Mo	$1.36 \cdot 10^3$	$1.98 \cdot 10^{-19}$	$1.90 \cdot 10^{-17}$	$1.46 \cdot 10^{-22}$	Mo
La	$0.36 \cdot 10^3$	$1.78 \cdot 10^{-10}$	$2.48 \cdot 10^{-8}$	$0.50 \cdot 10^{-12}$	La
Ce	$0.72 \cdot 10^3$	$0.39 \cdot 10^{-9}$	$0.55 \cdot 10^{-7}$	$0.54 \cdot 10^{-12}$	CeO,Ce
Eu	$0.36 \cdot 10^2$	$0.48 \cdot 10^1$	$0.72 \cdot 10^3$	$1.33 \cdot 10^{-1}$	Eu
Y	$1.24 \cdot 10^2$	$1.02 \cdot 10^{-9}$	$0.91 \cdot 10^{-7}$	$0.83 \cdot 10^{-11}$	Y
Zr	$1.22 \cdot 10^3$	$0.61 \cdot 10^{-18}$	$0.56 \cdot 10^{-16}$	$0.50 \cdot 10^{-21}$	Zr
Pu	$1.57 \cdot 10^4$	$2.65 \cdot 10^{-8}$	$0.65 \cdot 10^{-5}$	$1.69 \cdot 10^{-12}$	Pu
Np	$1.27 \cdot 10^2$	$0.76 \cdot 10^{-12}$	$1.80 \cdot 10^{-10}$	$0.60 \cdot 10^{-14}$	Np
U	$0.59 \cdot 10^5$	$2.75 \cdot 10^{-14}$	$0.65 \cdot 10^{-11}$	$0.47 \cdot 10^{-18}$	U, UI_2
Am	$3.12 \cdot 10^2$	$2.24 \cdot 10^{-3}$	0.54	$0.72 \cdot 10^{-5}$	Am
Cm	$2.16 \cdot 10^1$	$1.36 \cdot 10^{-8}$	$0.33 \cdot 10^{-5}$	$0.63 \cdot 10^{-9}$	Cm

---

[29] No retention in fuel and no adsorption or mechanical deposition on primary system structure.

4.4 Summary

The behavior and transport of radionuclides during a metal-fuel SFR core damage accident is complex. However, there has been significant past work undertaken to understand the phenomena involved, and historical reactor accidents and events provide real-world core damage results. Here, an assessment of the current state of knowledge is made, followed by a summary of past sodium reactor accidents and experiments.

Current State of Knowledge Summary

Table 4-11 provides a preliminary assessment of the current state of knowledge regarding radionuclide release and transport during a core damage accident. Each of the radionuclide groups is assessed in three categories:

Retention in & Release from Fuel Data – Data relating to retention of radionuclides within the fuel pin, and the form of any radionuclides released from the fuel pin into the primary sodium.

Solubility Data – Data relating to the solubility and chemical form of radionuclides in the primary sodium.

Vaporization Data – Data relating the likelihood of vaporization of radionuclides from the primary sodium pool into the cover gas region.

Each radionuclide group is assigned one of three knowledge levels dependent on the experimental results and models/correlations available for each of the three categories.

High – Detailed experimentation and/or correlations or theoretical models.

Medium – Experimentation but no models or correlations.

Low – Sparse experimental information.

The solubility data and vaporization data ratings are very similar for most radionuclide groups since they are thermodynamically linked, as explained in Appendix A.

Table 4-11: Preliminary Assessment of Radionuclide Transport Knowledge

Radionuclide Group	Retention in & Release from Fuel	Solubility	Vaporization
Noble Gases	High	High	High
Halogens	Medium	High	High
Alkali Metals	Medium	High	High
Tellurium Group	Medium	High	High
Alkaline Earths	Low	High	High
Noble Metals	Medium	Medium	Medium
Rare Earths	Low	Medium	Medium
Actinides	Medium	Low	Low

While the current state of knowledge for solubility and vaporization is high for most radionuclide groups, the assessment gives lower scores to the knowledge related to retention and release of radionuclides from the fuel pin. There are several reasons for this. First, the time-in-cycle and the

specific accident conditions can greatly influence if/how radionuclides are released from the fuel pin. Second, many past experiments have focused on the release of radionuclides from oxide fuel. While this data is still valuable for assessing how radionuclides behave in the sodium, it is less useful for understanding the release phenomena for metal fuel. As will be discussed in Section 6, it is possible that more information regarding in-pin behavior of radionuclides exists.

Past Accident/Experiment/Analysis Summary

There have been three past U.S. sodium reactor incidents involving metal fuel. These incidents provide insight into the behavior of metal fuel during real core damage events. Table 4-12 provides a summary of the three events, with releases into the cover gas highlighted in red and radionuclides found in the primary system highlighted in blue.

Table 4-12: Summary of Past Metal Fuel Sodium Reactor Incidents

Reactor	*SRE[1]*	*Fermi 1[2]*	*EBR-II[3]*
MW$_{th}$	20	100	62.5
Fuel Type	Unalloyed uranium metal and experimental elements	U-10 wt% Mo, Zr cladding	U-75%, Pu-15%, Zr-10% (experimental fuel element)
Neutron Spectrum	Thermal	Fast	Fast
Typical Core Outlet Temperature	538°C	427°C	482°C
Core Damage	Cladding failure of 13 fuel elements	Fuel melting in two assemblies, fuel damage in two others	Substantial melting of single experimental fuel element
Burnup of Damaged Fuel	0.1% Average	<0.05 a/o	Fresh Fuel
Radionuclide Inventory Released (Ci)	5,000 – 10,000	10,000	
Radionuclides Found in:			
Cover Gas	^{133}Xe, ^{85}Kr	^{133}Xe, ^{135}Xe, ^{85}Kr	^{133}Xe, ^{135}Xe
Primary Na	^{141}Ce, ^{144}Ce, ^{131}I, ^{103}Ru, ^{137}Cs, ^{134}Cs, ^{89}Sr, ^{90}Sr, ^{95}Zr-Nb, ^{140}Ba-La, ^{103}Ru	^{131}I, ^{137}Cs, ^{89}Sr, ^{90}Sr, ^{140}Ba-La	^{131}I, ^{133}I, ^{137}Cs
Primary Structure	^{89}Sr, ^{90}Sr, ^{95}Zr-Nb, ^{144}Ce, ^{137}Cs	^{141}Ce, ^{144}Ce, ^{133}I, ^{103}Ru, ^{95}Zr-Nb	^{137}Cs
Cold Trap	^{137}Cs, ^{125}Sb		^{134}Cs, ^{137}Cs

[1] [64] [81]
[2] [38] [64] [83]
[3] [64] [86] [87]

As can be seen, all three incidents involved some type of metal fuel. Also, the extent of fuel damage differed, ranging from cladding failure and eutectic melting at SRE, to repeated substantial melting of the experimental fuel element at EBR-II. Perhaps the biggest takeaway from the three incidents is that no radionuclides other than the noble gases of xenon and krypton were found in the cover gas region. This implies that significant retention occurred, whether in

the fuel or in the primary sodium, of many of the important radionuclides that are commonly a concern during LWR core damage accidents.

Several of the important insights from the three accidents, in conjunction with the RBCB and TREAT experiments, are summarized below:

- Metal fuel cladding breach results in a release of fission gases (including Xe and Kr) and bond sodium (which may contain Cs), but does not imply a release of fuel material.
- Metallic fuel does not react with the sodium coolant, limiting the likelihood of fuel damage propagation.
- The retention of radionuclides, other than the noble gases, in the fuel pin and primary sodium appears to be very high, (even during significant fuel melting and relocation, such as at Fermi 1 and the EBR-II experimental capsule).
- All three reactor incidents involved fairly fresh fuel with relatively low burnups.
- For all three reactor incidents, cleanup operations were conducted and the reactor resumed normal operation.

Lastly, the partial mechanistic source term assessment conducted as part of the ALMR project appears to show very little release from the sodium pool of any radionuclide other than the noble gases. Even though the complete core inventory was assumed to be released to the primary sodium at very high temperatures, the release fractions to the cover gas of some of the most volatile elements, such as cesium, were still on the order of 10^{-3}. If credit for in-fuel retention and plateout in the primary sodium is considered, these release fractions would be reduced even further.

5 Ex-Core Source Term Phenomena

Mechanistic source term phenomena associated with radionuclide releases originating from locations other than the core are reviewed in this section. This begins with an examination of a primary sodium fire, followed by a spent fuel handling accident, and lastly, releases from accidents involving cleanup systems are assessed. These accidents reviewed here are based on those found in [22] [24].

5.1 Primary Na Fire

Primary Sodium Fire Summary
The most likely primary sodium fire scenario is a maintenance error during shutdown that allows air to enter the cover gas region and interact with the primary sodium. Residual radioactivity from activated sodium and fission products from structurally failed fuel pins may still be present in the sodium during shutdown, which is usually held at low temperatures (~200°C). The sodium pool fire may continue until all available oxygen has been consumed. Past experimentation shows that the release of radionuclides from the pool is highly dependent on atmospheric conditions, the amount of sodium burned, and the temperature of the sodium burning.

Sodium fires will be a focal point of safety analysis for SFRs. In this case, the primary release pathway is associated with the transport of sodium activity and fission products (if fission products are present in the sodium coolant) from the primary reactor coolant to the containment atmosphere during a sodium fire event.

In the past, primary sodium fires were treated as part of the HDCA analysis, in which it was postulated that an energetic accident would result in major core damage as well as damage to the reactor vessel head, exposing the primary reactor coolant to the containment atmosphere. While it is assumed that HDCAs will not be considered within the licensing basis during a future MST licensing analysis due to the extremely small likelihood of their occurrence, the possibility of a primary sodium fire cannot entirely be eliminated.

A primary sodium fire scenario that may be found within the licensing basis would be the failure to secure an inert atmosphere before removing a plug in the reactor vessel head to perform primary pump maintenance/replacement during an outage. While argon gas is much denser than air at room temperature, at the higher temperatures in the cover gas region, argon density is close to that of the air in containment, which could result in air ingress if a penetration is available. In this scenario, the sodium pool is likely at relatively low temperatures (~200°C), as is typical during refueling or major maintenance. This is close to the self-ignition temperature of sodium pools in air [90].

Within the sodium pool, the presence of some fission products (from stochastic structural pin failures during normal operation) as well as activated sodium in the reactor coolant is likely. This section focuses on the transport of activated sodium and fission products discussed in Section 3.2.1. Section 5.1.1 will describe the current state of knowledge of the sodium burning process and of transport of sodium activity and fission products from a burning pool of sodium. Section 5.1.2 will provide a summary of relevant past analysis associated with the topic.

Figure 5–1 illustrates the primary sodium fire scenario described above, along with relevant transport phenomena. It is assumed that air ingress causes the sodium fire[30]. There are two mechanisms that can transport radionuclides from the sodium pool. First, radionuclides can be transported from the sodium directly by the sodium oxides that are formed during burning. Second, as the top of the sodium pool increases in temperature, vaporization of sodium and dissolved radionuclides will increase. Once in the cover gas region, the airborne radionuclides may mechanically deposit, but condensation is unlikely due to the elevated temperatures from the fire. The depth of the sodium pool makes sodium burning near the core unlikely, as the burning rate would first reduce, then likely self-extinguish, before reaching the core level due to a lack of oxygen and/or due to the for mation of a sodium oxide crust layer.
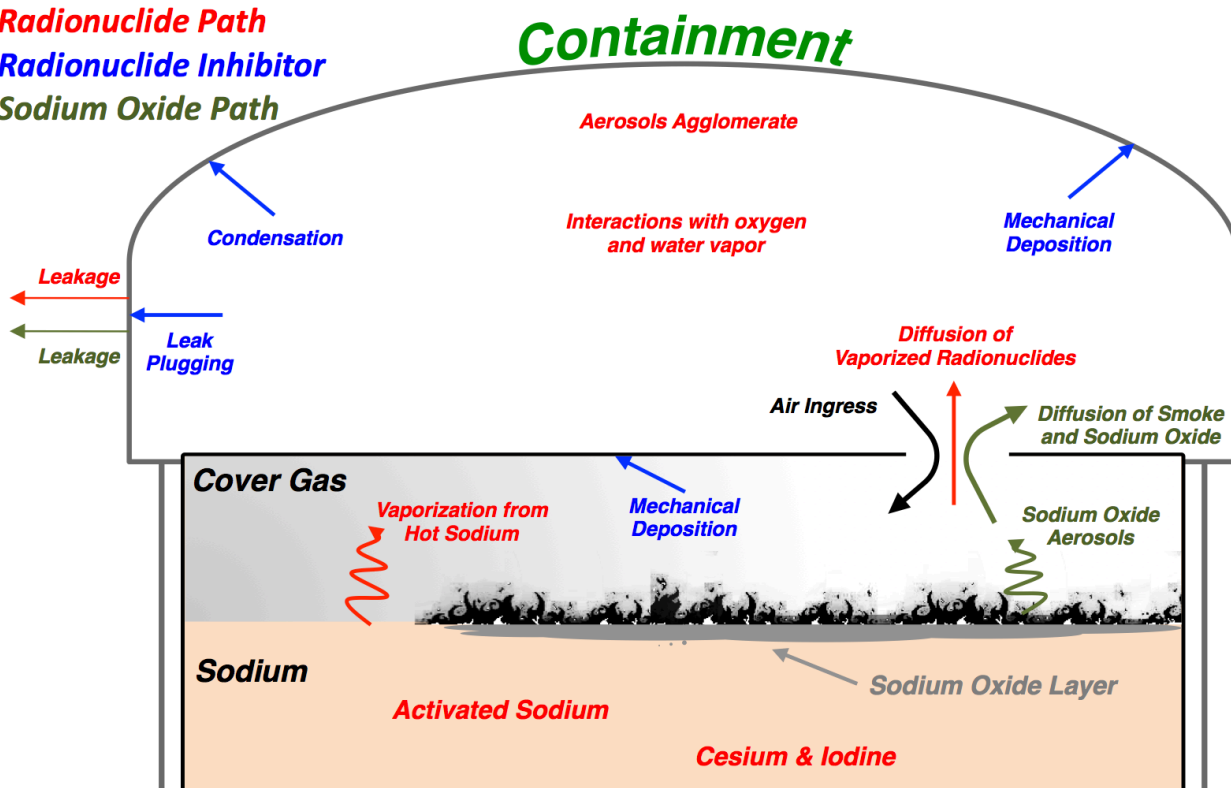
Figure 5–1: Primary Sodium Fire during Reactor Shutdown

With an open pathway to the containment, aerosols (*i.e.*, smoke, sodium oxides, and vaporized radionuclides) can diffuse out of the cover gas region (movement due to convection may also occur depending on the vessel head breach size and atmospheric conditions). Once in containment, aerosol may agglomerate, especially as interactions with oxygen and water vapor continue, leading to an increase in mechanical deposition and leak plugging. Condensation is also possible, provided relatively cold surfaces exist. The only barrier to radionuclide release is the

---

[30] It is assumed that self-ignition will occur, although this is not certain, as the self-ignition temperature of sodium in air is close to the shutdown temperature of the primary sodium pool [89].

containment structure; leakage according to the design basis is to be expected, although leakage may increase if pressure within containment rises due to the fire.

The inventory of the sodium pool, as described in Section 3.2.1, consists of activated sodium and fission products from structural pin failures during operation. Given that the accident occurs during shutdown, some of the activity will have decayed (especially the ^{24}Na, which has a short half-life). The behavior of these radionuclides is described in the following section.

5.1.1 Current State of Knowledge

Much of what is known about sodium pool fires is described in previous work by Newman [90] [91] [92] [93]. Newman explains that the sodium burning process initiates with flameless combustion involving rapid surface oxidations that result in the formation of a grey purple layer most likely comprised of surface reaction products close to the sodium surface. Sodium oxides are formed during this combustion process, and the location of their accumulation depends on the surface temperature of the sodium pool. At surface temperatures below 600°C, the sodium oxides accumulate atop the grey purple layer and are marked by yellow (sodium peroxide) and white (sodium monoxide) regions. The oxide surface layer thickness rapidly grows and wrinkles, with oxide nodules or pillars growing in random locations. Newman describes the beginning of vapor phase combustion occurring at pool temperature of 350°C to 450°C, marked by small flames with light and dark smoke emissions on the nodules. The oxide pillars are porous and act as wicks into which liquid sodium is drawn upward by capillary action and heated to temperatures exceeding the temperature of the bulk sodium beneath the wicks. At later stages in the combustion process, liquid sodium can begin to wet the oxide and is drawn up to react with peroxide releasing heat as it reduces the peroxide to oxide, further contributing to vapor phase combustion.

The vapor phase combustion on the nodules continues unless the sodium pool temperature reaches approximately 600°C, then the nodules appear to sink into the sodium pool or disappear. This behavior is attributed to wetting or decomposition of the grey purple layer, which provided support to the nodules.

Above 600°C, combustion occurs close to the liquid sodium surface as sodium oxide is formed and deposits on the sodium pool surface. These formations are wetted by the liquid sodium of the pool and sink as a result. This process continues and the oxide accumulates in the pool as more sodium oxide is formed on the surface and sinks as well. As the oxide accumulates in the pool, the burning rate of the sodium decreases. In fact, Newman states that sodium pool fires rarely burn to completion and a significant amount of unreacted sodium remains after the combustion process has ceased.

During a sodium pool fire, a fraction of the products of the combustion process, sodium oxides, leave the pool surface as smoke. The smoke fraction varies with temperature as shown in Figure 5–2 [91]. From 250°C to 600°C, the smoke fraction is relatively constant at approximately 0.11. Newman suggests that the unvarying smoke fraction in this temperature region is related to the transport of liquid sodium through the oxide wicks on the pool surface. Above 600°C, the oxide surface sinks into the pool as previously described and sodium vaporization occurs over the entire pool surface. This leads to a significant increase in the smoke fraction from 600°C to 750°C, with the smoke fraction being 0.18 at 650°C, 0.25 at 700°C, and 0.34 at 750°C. It should be noted that

the maximum temperature of an unconfined burning sodium pool with unlimited oxygen is stated to be approximately 730°C, which represents the steady state temperature with a balance between the heat generated during the combustion process and the heat losses to the sodium pool and the surroundings. In the experimental tests, sodium pool temperatures exceeding 730°C were maintained by external heating [91].
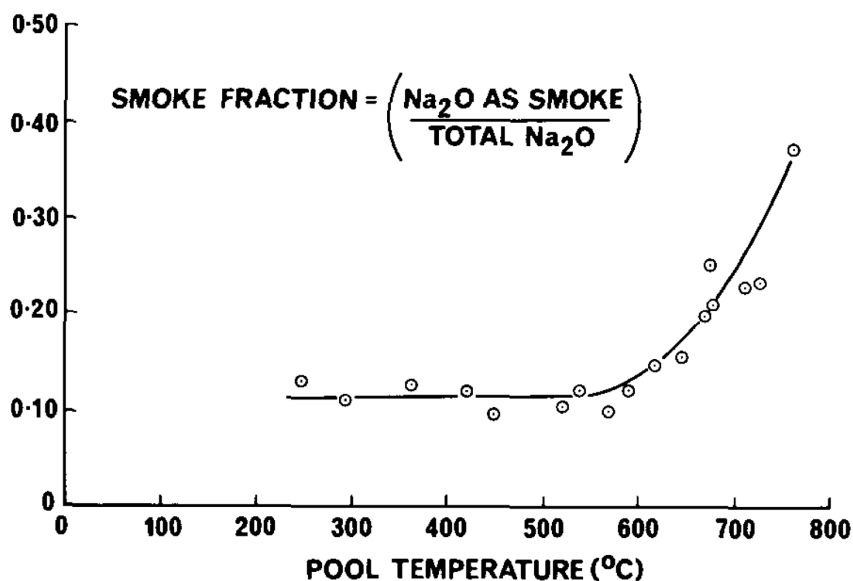
Figure 5–2: Sodium Fire Smoke Fraction Temperature Dependence [91]

As stated previously, the transport of sodium activity and fission products from the pool surface during a fire is the primary release pathway of interest in an MST assessment. Estimating the release fraction of each isotope in the sodium pool can be difficult because the release fractions vary with sodium pool temperature, atmospheric oxygen concentration, absolute humidity, and other scenario specific variables. The remainder of this section will describe the current state of knowledge of the transport during a sodium pool fire of certain isotopes (^{24}Na, ^{22}Na, ^{131}I, ^{137}Cs, and ^{134}Cs) that accounted for the majority of activity found in the primary sodium of EBR-II and FFTF subsequent to pin failures (See Figure 3–6).

The release fraction of activated sodium (^{24}Na and ^{22}Na) would be the same as the non-activated sodium smoke fraction. As mentioned previously, the release fractions are dependent on several variables, and previous experimental work by Sahoo et al. [94] and Kawahara et al. [95] have demonstrated different behavior than that shown in Figure 5–2[31]. The temperature dependency of the release fraction of sodium (and cesium) from experiments conducted by Sahoo et al. is shown in Figure 5–3 [94]. Around 400°C, the sodium release fraction is approximately 5% and monotonically increases to approximately 22% at 600°C. The results from Kawahara et al., shown in Figure 5–4 [95], are similar with a sodium release fraction of approximately 8% at 400°C and 25% at 600°C. It should be noted that ^{24}Na should not be a major concern for a postulated incident during an outage as it has a relatively short half-life (15 h).

---

[31] These experiments were conducted with small sodium samples, 0.5 - 120g.

The minor differences in the results between Sahoo et al. and Kawahara et al. can be attributed to the differences in oxygen concentrations and relative humidity of the test atmospheres. The testing by Sahoo et al. was conducted in an atmosphere with 21% oxygen and 70% relative humidity and the Kawahara et al. test atmosphere was 10% oxygen and 0% relative humidity. The dependency of the sodium release fraction on oxygen concentration and humidity were also investigated by Kawahara et al.; these results are shown in Figure 5–5 [95]. The authors of [95] attribute the decrease in the release fraction with increased oxygen concentration to the formation of a layer of sodium oxide scum on the surface of the test samples that impedes the vaporization of sodium.
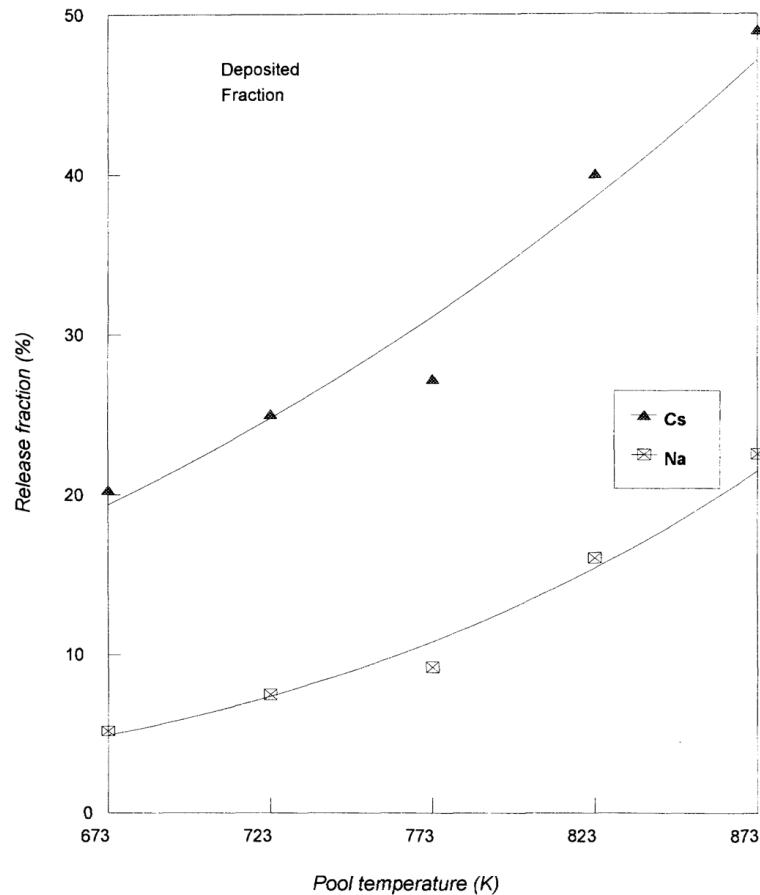
Figure 5–3: Sodium Fire Release Fraction versus Temperature [94]
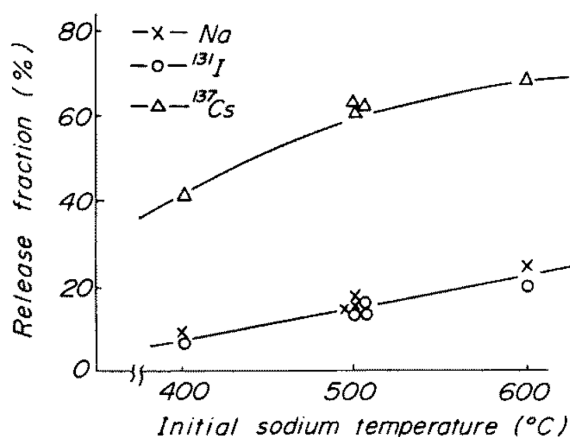
Figure 5–4: Sodium Fire Release Fraction versus Temperature [95]
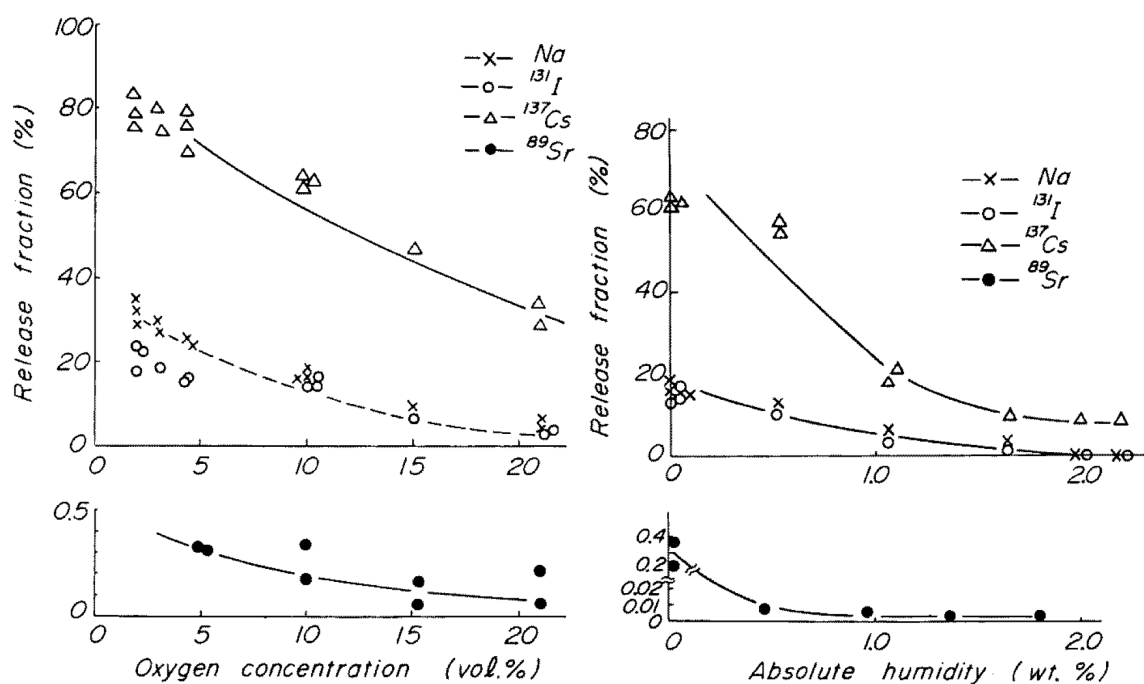
Figure 5–5: Sodium Fire Release Fraction Dependence on Oxygen and Humidity [95]

The release fraction of iodine from a sodium pool fire has been experimentally investigated in the past with significant work by Berlin et al. [96] and Kawahara et al. [95]. Berlin et al. performed four tests that demonstrated a time dependent iodine retention factor[32] from a burning sodium pool, as shown in Figure 5–6 [96]. In all of the tests, the iodine retention factor decreased as a function of time due to an increase in the sodium pool temperature from the fire. As the pool temperature increases, the vapor pressure of the iodine increases, leading to increased iodine vaporization. Mean retention factors of iodine reported by Berlin et al. range from 0.51 to 0.66 and final retention factors (at the end of the fire) range from 0.5 to 0.1. Tests by Kawahara et al. also demonstrate an increase in the release fraction of iodine as sodium pool temperatures increase from approximately 6% at 400°C to 20% at 600°C, as shown in Figure 5–4 [95]. While

---

[32] As per [95], the ratio between the concentration of the contaminant in the pool to that in the aerosol

Kawahara et al. suggests that the release fraction of iodine seems to be similar to that of sodium, Berlin et al. conclude that the emission of iodine is not proportional to sodium emission during a fire.
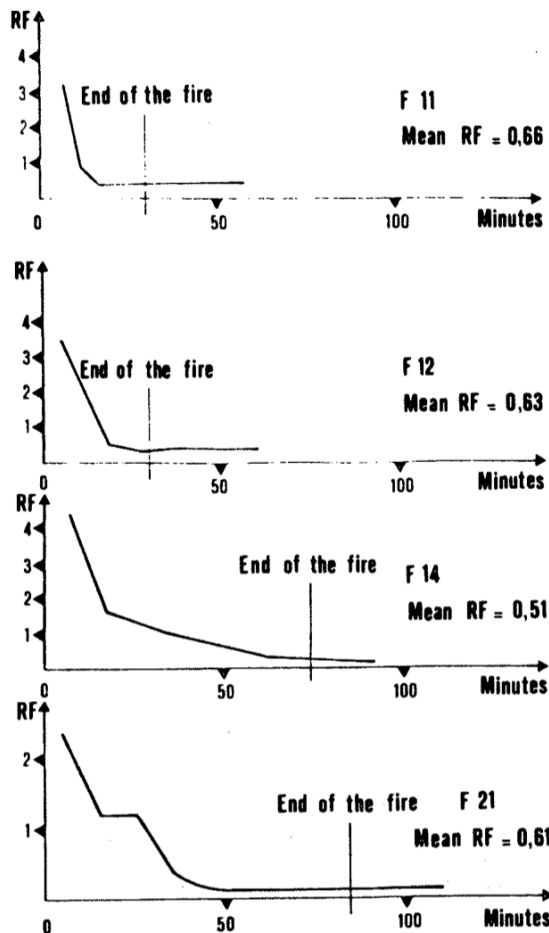
Figure 5–6: Sodium Fire - Time History of Retention Factors for Iodine [96]

Less experimental data are available on the release fraction of cesium (^{137}Cs and ^{134}Cs) during a sodium fire than that of sodium or iodine. However, both Sahoo et al. [94] and Kawahara et al. [95] have performed experiments to determine such a release fraction. The results of Sahoo et al. are shown in Figure 5–3 [94]. At 400°C, the cesium release fraction is approximately 20% and monotonically increases to approximately 49% at 600°C. The results from Kawahara et al. are shown in Figure 5–4 [95]. At 400°C, the cesium release fraction is approximately 40% and monotonically increases to approximately 70% at 600°C. Sahoo et al. attribute the differences between their experimental results and those reported by Kawahara et al. to differences in oxygen concentration and relative humidity in the test atmospheres. Sahoo et al. attributes the higher release fraction of cesium, compared to sodium, to its higher vapor pressure at a given temperature than sodium. It is important to note that the authors of these experiments only tested sodium temperatures up to 600°C. If they had performed experiments at higher sodium pool temperatures, they may have seen similar behavior to that observed by Newman [91] where the release fraction dramatically increased at sodium pool temperatures exceeding 600°C.

5.1.2 Past Analyses

The PRISM PSID [24] considered radionuclide release from a burning sodium pool as part of a larger HCDA analysis, which resulted in the loss of the reactor vessel head and exposure of the sodium pool to the upper containment atmosphere. This scenario is not analogous to the primary sodium fire during reactor shutdown outlined at the start of this section.

In the PRISM analysis, it was assumed that a leak path formed in the reactor vessel head allowing the cover gas to be released into the containment volume and air to enter the cover gas region, initiating a primary sodium fire. The fire was assumed to continue until all of the oxygen in the containment dome was consumed. The releases attributed to the fire are as follows: 0.4% of the sodium, 0.8% of the halogens, 1.6% of the alkali metals, 0.004% of the tellurium and ruthenium, 0.0016% of the strontium and barium, and 0.0008% of the fuel and other fission products.

5.2 Spent Fuel Handling Accident

Spent Fuel Handling Accident Summary
A gas-cooled moving cask incident would likely result in the release of some of the spent fuel's fission gases, including the noble gases, if the cask barrier were to be breached, cooling capability lost, and pin failure to occur. A sodium pot moving cask could present a more serious accident if a breach occurred large enough to result in a sodium fire. The sodium fire could potentially result in larger releases from the spent fuel, but this scenario is heavily dependent on the design, and should be addressed by a vendor pursing the sodium pot moving cask option. Both accidents could occur outside of containment, as spent fuel is moved to long-term storage areas.

As discussed in section 3.1.3, there are several methods for transporting spent fuel from the reactor vessel to a storage location. Of interest here are the gas-cooled moving cask and sodium pot moving cask. The sodium pot transfer shoot option is not reviewed here, as the shoot is usually directly connected to the reactor vessel, and the spent fuel never leaves an inerted atmosphere. This section reviews the radionuclide transport phenomena associated with an accident involving both types of external moving casks. The gas-cooled moving cask incident is examined first.

Gas-Cooled Moving Cask

The benefit of the gas-cooled cask is the limited sodium inventory involved with the transfer of spent fuel, as the spent fuel will only be wetted with residual sodium from the reactor vessel. However, the thermal conductivity of the inert gas is much lower than sodium, which complicates cooling of the spent fuel. For this accident, it is assumed that the gas-cooled cask seals fail, providing a pathway from inside the cask to the environment (as the cask may be outside of containment). The accident is coupled with a loss of cask cooling, likely as a consequence of the event that led to seal failure. This leads to overheating and the failure of some fraction of the pins in the spent fuel assembly. This scenario is illustrated in Figure 5–7. While the cask is shown inside containment, it may also be in an auxiliary building.

As the cladding fails, fission gases would be released, including the noble gases. Other volatile substances may also vaporize and escape, such as cesium that was present in the bond sodium. The nature of the cask leak plays an important role for other phenomena. If the cask leak is small, the assembly may remain in an inerted environment. However, if the leak is large enough to cause air to intermix in the cask, oxidation of the residual sodium may begin. Unlike a release to the cover gas in the reactor vessel, the presence of oxygen and water vapor may create more or less volatile forms of some of the radionuclides.

As for the inventory of radionuclides, it is important to recall that the spent fuel assemblies have likely decayed in the reactor vessel for many months. This will greatly reduce the inventory of the noble gases, which have relatively short half-lives.
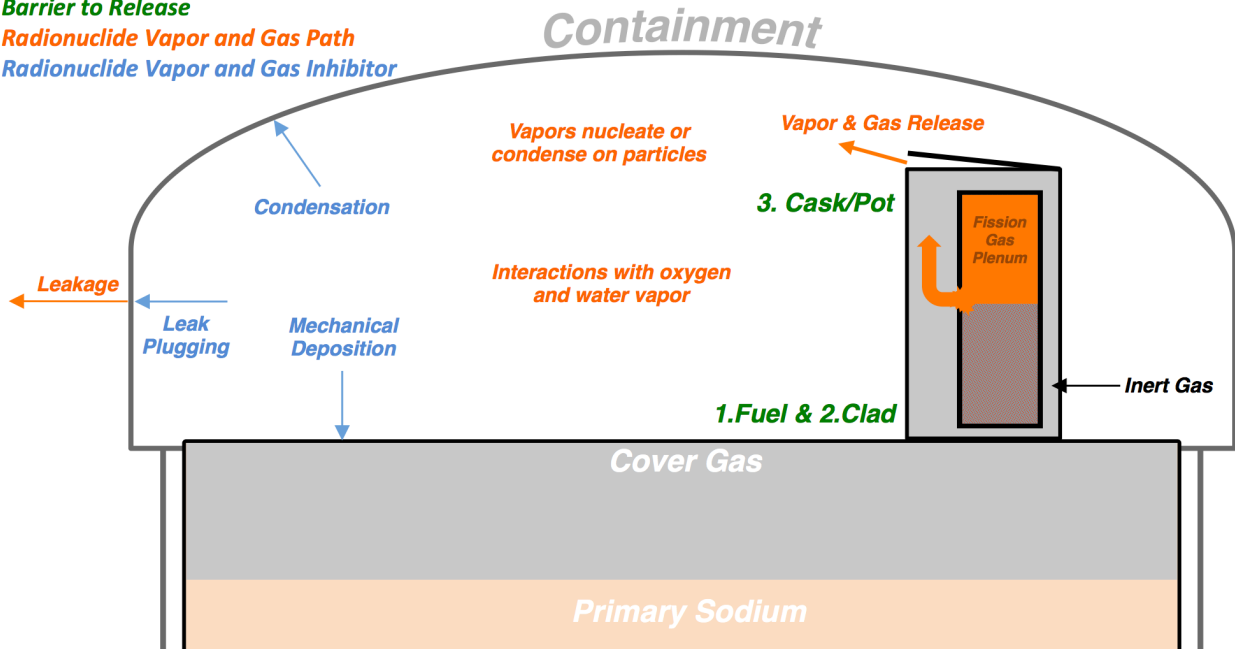
Figure 5–7: Spent Fuel Handling Accident – Gas-Cooled Moving Cask

Sodium Pot Moving Cask

The second scenario involves a sodium pot moving cask accident. Here, the spent fuel elements are placed into a pot containing sodium. While the higher thermal conductivity and heat capacity of the sodium aid in heat removal, the possibility of a sodium fire with a cask leak is greater. That event is likely the worst-case scenario, and is examined here.

If a breach of the cask occurs, air could enter the cask. Sodium does not self-ignite in air at low temperatures, but if temperatures above ~200°C are experienced, self-ignition is a possibility [90]. If a fire does ignite, the fuel pins could be subjected to high temperatures, which could result in pin failures due to overheating. The radionuclides that escape the pins are likely to be quickly vaporized due to the fire. It is unclear what additional releases would occur as the fuel assembly is engulfed in the sodium fire, as sodium fire flame temperatures can exceed 1700°C [97].
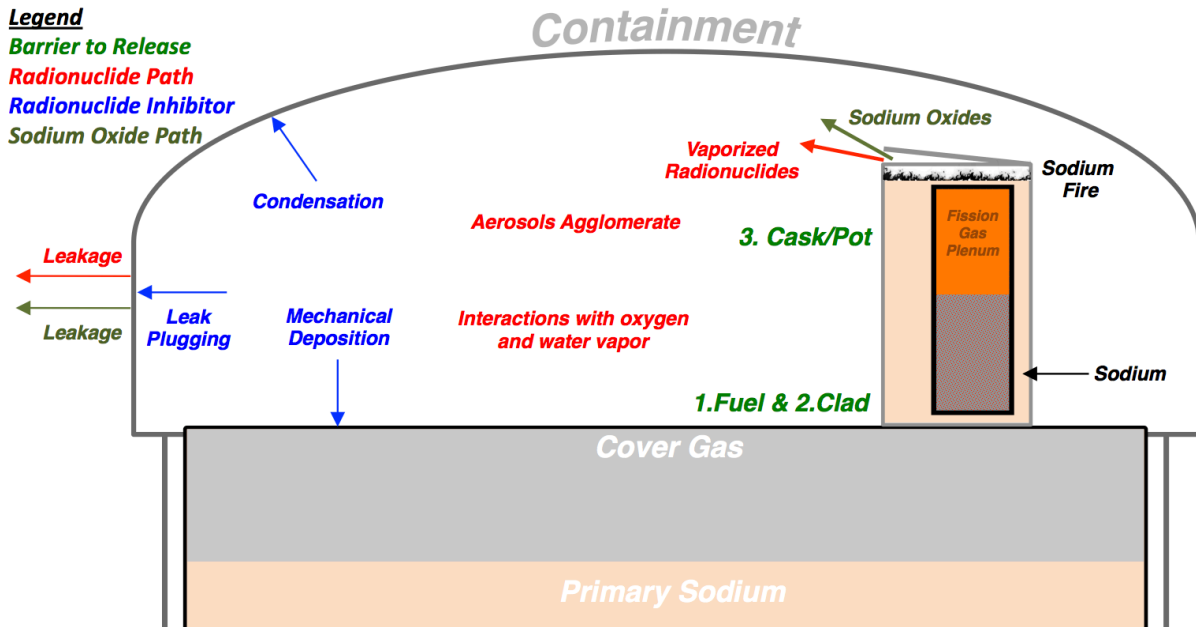
Figure 5–8: Spent Fuel Handling Accident – Sodium Pot Moving Cask

5.2.1 Current State of Knowledge

The phenomena associated with the transport of radionuclides from the gas-cooled moving cask incident are well understood. Noble gases will be released from the cask and will be transported to the environment. Other fission gases and vapors would also be released, and may interact with the oxygen and water vapor in the environment. In contrast, the phenomena associated with a sodium pot moving cask accident, and subsequent sodium fire, may add much more complexity.

If a sodium fire were to occur in a sodium pot moving cask due to a significant breach, the spent fuel assembly could be subjected to very high temperatures. Fuel melting could result in the release of additional radionuclides. As described in Section 4.2, metal fuel was subjected to very high temperatures in crucible tests to gauge the release of radionuclides during reprocessing. This information provides some insight into the scenario, but temperatures even higher than these tests (which took place at ~1300°C) may be possible. However, information on the burning of uranium and plutonium is available from past experimentation [98] and accidents [54]. If a designer were to pursue a sodium pot moving cask option, additional analyses would likely be needed to gauge the plausibility of such an accident and its possible consequences.

5.2.2 Past Analyses

A fuel transfer cask cover gas release was examined in the PRISM PSID [24]. In this analysis, it was assumed that three spent fuel subassemblies had been in storage within the reactor vessel for 20 months before being loaded into the transfer cask. The reactor refueling temperature is ~200°C, and while the transfer cask is passively cooled, fuel inside could reach ~400°C. For the accident scenario, a total of five fuel pins in the three subassemblies fail due to this increase in temperature[33]. The failed pins release fission gases and volatile inventory into the transfer cask. The valves on the cask are assumed to fail and do not seal properly, which allows leakage from

---

[33] The mechanism of pin failure at 400°C is not stated.

the cask as it becomes pressurized due to the heating of the inert cover gas. A leak rate of 5% per day is assumed. Only krypton, xenon, and iodine were analyzed. Very small doses were calculated at the exclusion area boundary and low population zone.

The NRC, in its review stated [23]:

> The requirement of leaving spent fuel assemblies in the reactor vessel for almost 2 years makes handling the spent fuel less likely to result in releases due to fuel overheating. This also makes many aspects of the fuel handling easier. The method of moving the spent fuel between the reactor vessel and the fuel cycle facility by a self-propelled cask transporter should minimize the exposure to plant personnel.

5.3 Primary Sodium Cleanup System Accident

> **Primary Sodium Cleanup System Accident Summary**
> *The most severe primary sodium cleanup system accident is likely a breach in the cesium or cold trap, followed by a sodium fire. Both traps are expected to be located in a room or vault designed to contain sodium fires (air tight to limit oxygen, steel-lined floors and walls). The fire could potentially vaporize radionuclides found in both traps, but the extent of the fire and radionuclide release is highly dependent on the size/design of the traps and the room. Since primary sodium cleanup systems have historically been located within the upper containment, another barrier to environmental release is provided.*

As described in Section 3.2.2, the main purpose of primary sodium cleanup systems is to remove the impurities of oxygen and hydrogen during normal operation to prevent component corrosion and activation. However, these cleanup systems can also capture some the radioactive inventory of the primary sodium. This section examines the transport phenomena related to an accident involving the primary sodium cleanup system.

There are several design options for the primary sodium cleanup system. Some operate continuously during reactor operation, whereas others are only run during shutdown. The cold trap can be located within the reactor vessel, or placed in a room within containment. Also, a dedicated cesium trap may be included (using vitreous carbon), located near the cold trap.

For this analysis, a system similar to the one used at EBR-II is examined. Here, the purification system is run continuously, and the cold trap and dedicated cesium trap is placed in a room within containment. The cold trap and cesium trap are also located within a vault, which is a room designed to accommodate sodium leaks and fires. Some designs have proposed the use of an inerted room atmosphere for the primary sodium cleanup system as a method to prevent sodium fires.

The layout of the system and the associated release barriers are illustrated in Figure 5–9. The location of the system within containment is usually above the reactor head in an annular region. The most challenging accident scenario would likely involve a leak in the cesium trap, cold trap, or associated piping that would result in a sodium fire. This could result in the vaporization of a portion of the radionuclides in both traps and within the primary sodium itself. The transport phenomena are not listed in the diagram, as they are similar to the sodium fire accident reviewed in Section 5.1.

The behavior of the fire depends on several factors. First, the design of the vault that contains the traps could limit the amount of oxygen available for the fire. Also, the use of steel liners on the floors and walls could prevent any sodium-concrete interactions. Second, the size of the cold trap, cesium trap, and piping in the room would determine the amount of sodium available for the fire. For example, EBR-II had a 500 gallon cold trap, while the PRISM design proposes a 1000 gallon trap [24]. Also, some designs contain multiple traps, so that one can be serviced while the reactor is operating.
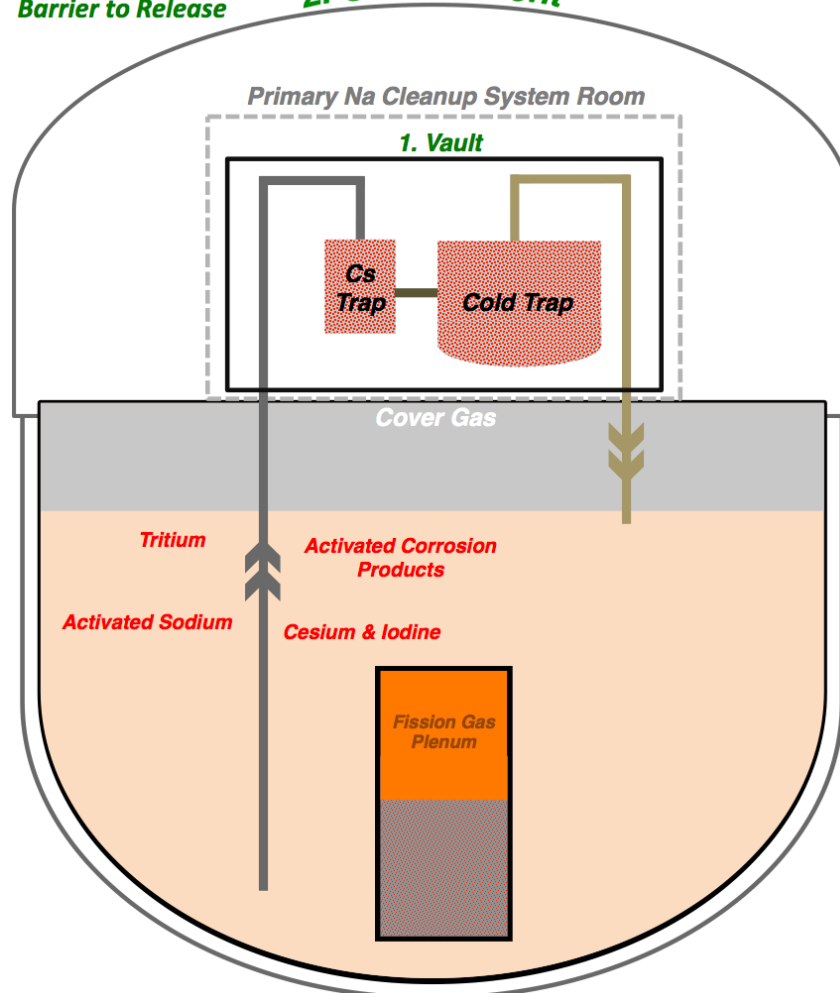
Figure 5–9: Primary Sodium Cleanup System Accident Scenario

The phenomena associated with the transport of radionuclides during a sodium fire are discussed in Section 5.1. Within the cold trap, the radionuclides will consist largely of activated corrosion products, with some tritium and cesium present. The majority of the cesium inventory will be held within the cesium trap, and the exact quantity will depend on the number of pin failures, if any, during that operational run[34]. Activated sodium will be present in both traps, and the associated piping.

One of the most important factors for this scenario is whether the radionuclides and fire can be contained within the vault. If so, it would limit the release to the larger containment volume, where leakage could occur to the environment. This depends on several factors, including whether the vault can contain the pressure rise that would occur due to a fire before it consumes all available oxygen and self-extinguishes. These factors should be considered as part of the vault design process.

---

[34] Stochastic structural pin failures during operation, not pin failures due to accident conditions.

5.3.1 Current State of Knowledge

The knowledge necessary to analyze this scenario is nearly identical to the primary sodium fire discussed in Section 5.1. The only differences are considerations of the inventory within the cold and cesium traps, and whether this inventory would be available for transport during a fire. In the absence of a completed primary sodium cleanup system design, it is difficult to mechanistically predict the consequences. However, since the cold trap works by decreasing the temperature of the sodium below the saturation point of the contaminants, it is likely appropriate to assume that the rise in temperature caused by the fire would mobilize many of the radionuclides. On the other hand, the cesium trap works simply by the filtering action of vitreous carbon. Whether this carbon would burn is dependent on the grade of the vitreous carbon used for the trapping.

5.3.2 Past Analyses

The PRISM PSID examined a release from a cold trap accident to the environment during an outage[35] [24]. During shutdown, the primary sodium purification system would be operated to remove impurities from the primary circuit. The accident analysis assumed that two pins had failed during operation, and that all fission products as well as 0.01% of the transuranics were uniformly distributed in the primary sodium. Activation of the sodium to form ^{24}Na and ^{22}Na was also considered. The postulated accident involved a spill from the 1000-gallon cold trap, and subsequent fire with fractional airborne releases.

The NRC, in its review, stated [23]:

> Unless the PRISM metal-fuel performance (fuel pin failures in an equilibrium core) is far worse than expected, this event is unlikely to result in a major release to the environment. As the design matures, this event should be examined further, particularly with respect to possible damage to safety systems resulting from sodium fires. It should be noted, however, that if the designers of PRISM incorporate state-of-the-art sodium fire detection and mitigation systems, the likelihood of a problem in this area should be minimized.

---

[35] The PRISM design only conducted primary sodium and cover gas purification operations during shutdown, not normal operation.

5.4 Cover Gas Cleanup System Accident

> **Cover Gas Cleanup System Accident Summary**
> For the reference design considered here, the most severe cover gas cleanup system accident is likely a rupture of the cryogenic distillation column, coupled with a loss of cooling. This would result in the revaporization of the noble gases stored inside. The inventory of radionuclides would depend on the number of structural pin failures and the decay time. Since the cleanup system is likely outside the containment, there are few engineered barriers to release. However, the short half-lives of many of the noble gas radionuclides may greatly reduce the release.

As described in Section 3.2.4, the main purposes of the cover gas cleanup system are to remove radionuclides from the cover gas region (to avoid radionuclide leaks into containment and to facilitate maintenance) and to identify ruptured fuel pins through the use of a gas tagging system. Like the primary sodium cleanup system, there are varying designs, with some SFRs using a continuous operation strategy, while others clean the bulk cover gas during shutdown.

This analysis will assume a system similar to the one used at EBR-II, which was utilized during reactor operation for both cover gas purification and monitoring for the fuel pin gas tagging system. The system used a cryogenic distillation column, which used very low temperatures to condense and remove the noble gases. The distillation column was placed within a cold box, which was designed to contain the radionuclides were a leak or rupture to occur. The location of the system outside of containment is also an important factor, as a traditional release barrier is therefore unavailable.

A layout of the example system, with barriers highlighted, can be seen in Figure 5–10[36]. Here, the greatest challenge to the system would simply be an increase in temperature due to a loss of the cooling system in conjunction with a rupture or break, resulting in a release of the noble gases stored in the distillation column. This includes xenon and krypton from failed fuel pins[37], activated argon, and radioactive neon, which is a decay product of activated sodium. Since these are all noble gases, they have extremely high vapor pressures and are essentially unreactive with other elements/substances. A rupture to the system, which causes an increase in temperature and vaporization of the noble gases, would likely lead to the release of the majority of the distillation column inventory.

While the distillation column may be located within a cold box, some leakage could occur even if the cold box functions as designed. Since the system is likely located outside of containment, there would likely be no other barriers of sufficient resistance to impede the release of the noble gases to the environment. The quantity of radionuclides within the system is likely the most important factor, and would depend on the time-in-cycle, the number of fuel pin ruptures (and time since the ruptures), and length of reactor operation to that point. As with the release of the noble gases from the core during an accident, since many of the noble gases have short half-lives,

---

[36] The transport phenomena are not show in the diagram, as the transport of noble gases is fairly direct, with little interaction with other elements/compounds.

[37] Stochastic structural pin failures during operation, not pin failures due to accident conditions.

the decay that occurs within the distillation column itself, coupled with a delay in release due to holdup in the cold box, could play a significant role in reducing the radionuclide inventory.
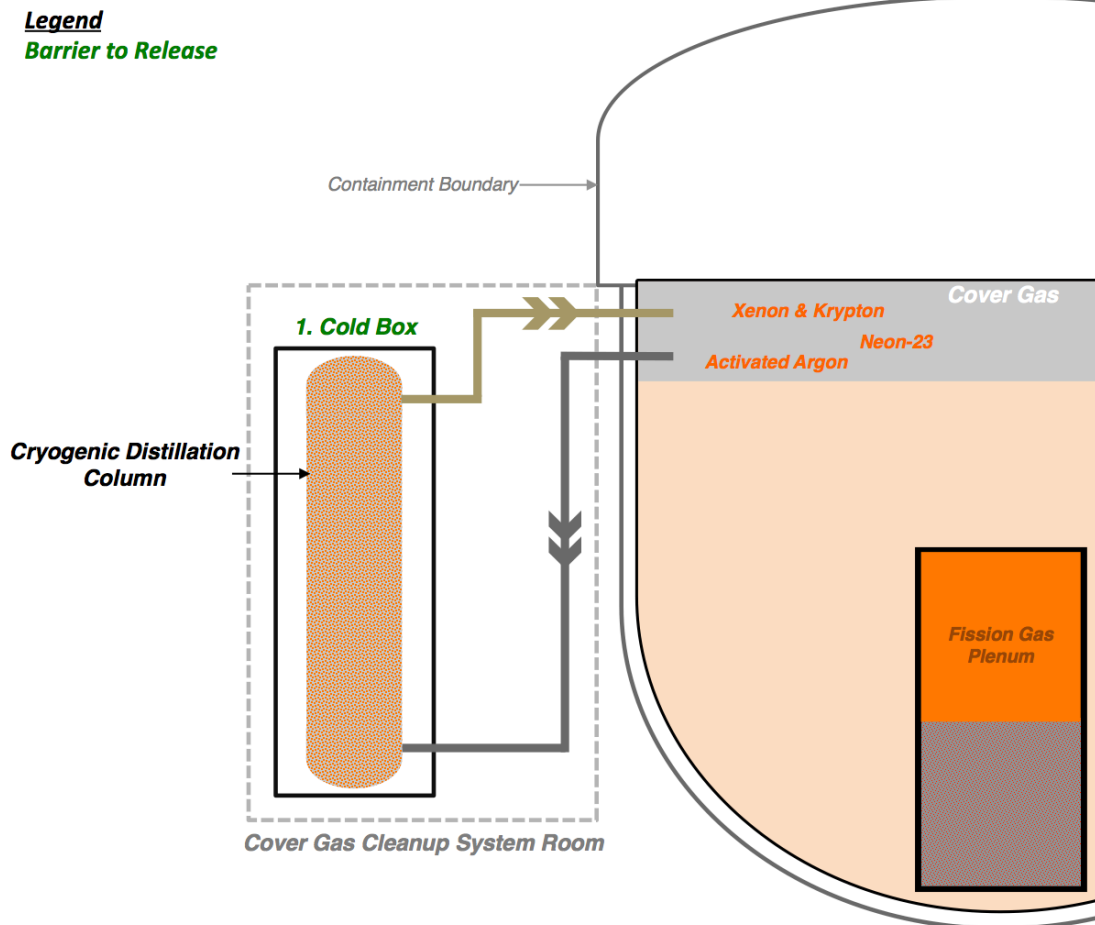
Figure 5–10: Cover Gas Cleanup System Accident Scenario

5.4.1 Current State of Knowledge

Since noble gases will vaporize at temperatures slightly above absolute zero and are highly unreactive, the single-phase transport of noble gases is fairly straightforward to assess. Therefore, the only knowledge needed for this analysis is the quantity of radionuclides in the system (scenario dependent) and the design of any cold box or similar vault (design dependent).

5.4.2 Past Analyses

As with releases from the cold trap, past SFR licensing efforts have also investigated the possible release from the cover gas cleanup system. The PRISM PSID examines a release during an outage[38] from the cover gas cleanup vehicle directly to the environment [24]. The analysis assumed that three fuel pin failures had occurred since the previous outage and cover gas purification efforts. Only the gaseous fission products of xenon and krypton were considered for the analysis.

---

[38] The PRISM design only conducted primary sodium and cover gas purification operations during shutdown, not normal operation.

The NRC, in its review, stated [23]:

> As long as it can be shown that only noble gases are involved in a release of the cover gas, this event is not expected to be a major safety concern. Release of the cover gas is likely to result in a small release of radioactive gases.

6 Assessment of Tools and Gaps

The following two sections assess the ability to model the phenomena outlined in Sections 4 and 5, and review gaps in both the knowledge base and modeling capabilities in the context of site suitability and LBE evaluations.

6.1 Modeling Capabilities

6.1.1 Core Damage and Radionuclide Transport

This section reviews the modeling capabilities related to core damage accidents. For each step of the calculation, available tools and their technical adequacy are discussed. Comments on the possible use of these codes in a licensing application are provided in the final part of this section.

> ### What radionuclides are present in the fuel?

Depletion codes, such as ORIGEN [99], are readily available and sufficient for determining what radionuclides will be present in the fuel at the time of an accident.

> ### What are the chemical forms of the radionuclides in the fuel?

Equilibrium thermodynamics codes, such as HSC Chemistry [100], can provide insight into the chemical forms of the radionuclides within the fuel (such as whether CsI will form in the fuel). However, there are drawbacks to equilibrium calculations. Thermodynamic equilibrium codes assume that each radionuclide will form the most thermodynamically preferred compound (based on chemical potential and Gibbs free energy, as explained in Appendix A). However, as discussed in Section 4.2.2, even though the formation of CsI is the thermodynamically preferred compound of iodine, and cesium atoms outnumber iodine atoms, the fact that there are many orders of magnitude more atoms of uranium in the fuel will likely lead to the creation of some UI_3 before those iodine atoms come into contact with a cesium atom. While equilibrium codes provide insight into the forms radionuclides may take, they do not necessarily tell the complete story. This drawback is likely not something that can be addressed through additional modeling capabilities, but insight can be gained through the analysis of previously irradiated fuel.

> ### What is the condition of the fuel prior to the accident?

For zirconium-based metal fuels, the Argonne fuel performance code LIFE-METAL [101], which is currently being updated, is capable of assessing the condition of the fuel prior to the occurrence of an accident. LIFE-METAL is the metallic fuel version of the LIFE code series, and began development in the early 1990s as part of the ALMR project. It is a steady-state fuel behavior code that is capable of assessing the swelling of the fuel due to the buildup of fission gases and determining the resulting cladding strain. This information is important for determining whether cladding will fail during accident conditions.

> ### How does the fuel behave during the accident?

SAS4A/SASSYS-1 [102] has historically been used by SFR researchers and vendors for analyzing SFR operational transients and accidents. It includes mechanistic models of steady-state and transient thermal, hydraulic, neutronic, and mechanical phenomena. The purpose of SAS4A/SASSYS-1 is to quantify the consequences of SFR severe accidents in terms of core damage. Of particular interest here are the code's models for metal fuel failure and relocation. There are multiple modules in the code for tracking cladding integrity, fuel movement (in- and ex-pin), and fuel/clad mechanical interaction. The SSCOMP module provides an empirically-developed database of metal fuel material properties and pre-transient characterization, including treatment of in-pin material migration caused by diffusion. Pre-failure pin behavior is described using the DEFORM-5 and FPIN2 modules; fission gas release, eutectic formation and clad strain are assessed in these modules to determine the timing and location of pin failure. The DEFORM-5 module, based on the robust, oxide-specific models of DEFORM-4, is integrated with the fuel relocation modules, PINACLE and LEVITATE, but contains limited phenomenology as it was primarily developed when knowledge of metal fuel transient behavior was immature. The FPIN2 module contains rigorous, mechanistic models of metal fuel transient behavior based on the current state of knowledge, but has not yet been fully integrated with the fuel relocation modules. In-pin fuel relocation is modeled using the PINACLE module, while ex-pin fuel movement and pin-to pin failure propagation are treated using LEVITATE. While the metal fuel-specific modules are intended to treat the phenomenology unique to metal fuels, some models are still based on oxide fuel behavior. However, some validation efforts for the metal fuel models have been completed based on the M-Series tests in TREAT [35].

> ### What radionuclides are released from the fuel during the accident?

SAS4A/SASSYS-1 can be used to assess the movement of fuel both before and after cladding has failed, but it does not estimate the radionuclide release from fuel. There appears to be no current modeling capability available to address this question. As stated in Section 4.4, the state of knowledge regarding radionuclide releases from metal fuel is less refined than the knowledge associated with radionuclide behavior in the sodium. Part of this gap is due to the focus on oxide fuel for many past fuel release experiments.

> ### How do the released radionuclides interact with sodium?

As with radionuclide formations in the fuel, thermodynamic equilibrium codes can be used to gain insight into the behavior of radionuclides in the primary sodium. However, once again there are drawbacks as the codes will assume complete mixing for the equilibrium calculation. This assumption is likely to overestimate retention within the primary sodium, where perfect mixing is unrealistic.

Other codes have been developed specifically to investigate the behavior of radionuclides in sodium. In the 1990s, the Japanese code TRACER [103] (Transport Phenomena of Radionuclides for Accident Consequence Evaluation of Reactor) was created to analyze the release of

radionuclides from the fuel into the sodium, transport through the sodium to the cover gas, and transport within the cover gas region. Figure 6–1 describes TRACER's analytical models. TRACER does not analyze the solubility of radionuclides in sodium or retention within sodium; instead the code focuses on bubble transport. Part of the reasoning behind this modeling choice is the prevalence of oxide fuel for SFRs in Japan, which is more likely to experience high temperatures and vaporization during accident sequences. The last revision of the TRACER code was released in 2005, and the code is no longer under development.

Behavior	Analytical model
1. Fuel Pin Temperature	•Input (function)
2. FP release from fuel to coolant	•Direct release (NUREG-0772) •Release from gas plenum •Specify initial bubble size based on gas ejection velocity •Input for gas composition
3. Coolant flow	•Flow network model
4. Bubble behavior a) Components b) Shape c) Size change d) Velocity e) Heat transfer f) EOS g) Diffusion between gas and liquid phases h) Deposition as aerosol	•Condensable, Noncondensable, Aerosol •Sphere, Ellipsoidal, and Spherical-cap •Variation due to heat transfer, head, and break-up based on instability •One dimensional model •Convection and Radiation •Ideal gas •Mass transfer coefficient in both phases •Considered
5. FP transfer model into cover gas	•Diffusion •Equilibrium evaporation
6. Deposition onto wall	•Considered

Figure 6–1: TRACER Analytical Models [104]

Other models have also been developed to track bubble progression within the primary sodium, but the focus of these codes is on HCDAs in oxide cores, which would create a large bubble containing vaporized fuel and fission products [105].

In the U.S., bubble transport computer models were developed as part of the ALMR project in the 1990s. However, these models would likely need updated for current use since they have not been maintained.

> What radionuclides will transport
> to the cover gas region?

Equilibrium thermodynamics can again provide insight into the vaporization of radionuclides from sodium pools. Bounding estimations of radionuclide release to the cover gas can be made using the gas-liquid equilibrium constant K_d (discussed in Appendix A), which does not require sophisticated computer models.

The REVOLS (Release of Volatile Species) code was developed at the Nuclear Research Center in Karlsruhe Germany in the early 1990s [69]. The code included a more detailed thermodynamic

model of the evaporative releases of radionuclides from sodium pools utilizing the Gibbs free energy and activity coefficients of the radionuclides (these topics are discussed in Appendix A). Figure 6–2 shows the input requirements and output data from a REVOLS calculation. REVOLS was designed to be coupled with CONTAIN-LMR, which is discussed in the following section. No information has been published regarding REVOLS since 1994, and it is assumed that the code is no longer under development or supported.
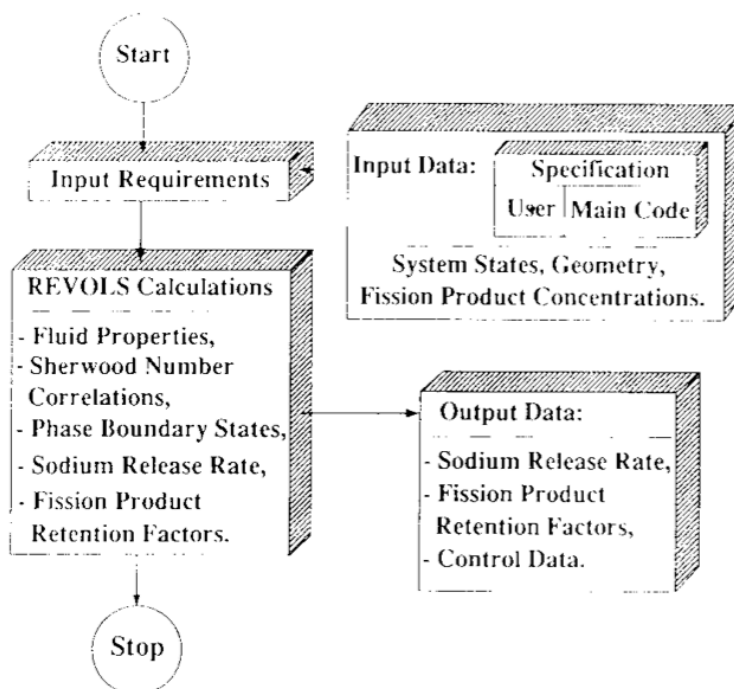
Figure 6–2: REVOLS Computational Flow Chart [106]

> *How will the radionuclides behave in the cover gas region and containment?*

Radionuclide behavior in the cover gas region and in containment can be modeled using computational tools such as CONTAIN-LMR [107] or MELCOR [108]. Both codes use aerosol models derived from the MAEROS code [109], which requires the user to specify the radionuclides present and aerosol size distributions for each radionuclide. Both CONTAIN-LMR and MELCOR model agglomeration processes such as Brownian, gravitational, and turbulent motion. Four deposition processes are treated in the codes: gravitational settling, diffusiophoresis, thermophoresis, and particle diffusion. In MELCOR, phenomena such as adsorption, chemisorption and other important chemical reactions can be simulated for radionuclides, but the processes are controlled entirely by user-specified parameters. While these tools exist, their performance is dependent on the state of knowledge available to the user. It is important to also note that, in an effort to reduce computational complexity, or due to a lack of phenomenologically consistent models, these tools may omit certain transport and deposition effects from the models that may have important effects on the associated phenomenology.

> *How will the radionuclides behave in the environment?*

Gaussian dispersal models are the nuclear industry standard for the modeling of radionuclide transport in the environment. Gaussian dispersal codes such as WinMACCS/MACCS2 [110] are readily available and sufficient for the source term calculation needs.

A brief summary of the current modeling capabilities is given in Table 6-1. The capabilities are considered adequate if the software can calculate all necessary information for that stage of the analysis. Some tools have partial capabilities, like HSC, since they can either provide general insight or address part of the problem, but they may not be able to provide the complete calculation. It should be noted that even though a capability may not be adequately fulfilled, an assessment of radionuclide transport is not precluded. For example, using gas-liquid equilibrium constants, it is possible to determine bounding values of radionuclide transport to the cover gas even though a code to complete this specific calculation for SFR accidents does not exist.

Table 6-1: Summary of Current Computer Technical Modeling Capabilities

Stage of Analysis	Computer Models	Adequate Technical Capabilities
Radionuclides in Fuel	*ORIGEN 2.2*	*Yes*
Radionuclide Chemical Form in Fuel	*HSC (thermodynamic equilibrium code)*	*Partial*
Condition of Fuel Prior to Accident	*LIFE-METAL*	*Yes*
Fuel Behavior during Accident	*SAS4A/SASSYS-1*	*Yes*
Radionuclides released during Accident	*SAS4A/SASSYS-1*	*Partial*
Radionuclide Behavior in Sodium	*HSC (thermodynamic equilibrium code)*	*Partial*
Radionuclide transport to Cover Gas	*HSC (thermodynamic equilibrium code)*	*Partial*
Radionuclide Behavior in Cover Gas/Containment	*CONTAIN-LMR/MELCOR*	*Partial*
Radionuclide Behavior in Environment	*WinMACCS/MACCS2*	*Yes*

It is important to note that if the computer codes described above are to be used for independent safety review by the regulator, they may need to reviewed and endorsed for use by the NRC before they can be considered adequate for licensing. Whether the computer codes listed would need additional qualification or validation depends on the exact use of the code within the SFR licensing application and also the MST qualification requirements from the NRC (which is considered an open issue and discussed in greater detail at the end of Section 6.2).

6.1.2 Sodium Fire

This section reviews the modeling capabilities related to sodium fire accidents. Sodium fires could be modeled using existing computational tools such as SOFIRE II [111] or CONTAIN-LMR [107]. SOFIRE II was developed by Rockwell International to describe the pressure and temperature history of the containment atmosphere following a postulated sodium leak/spill leading to a sodium pool fire. The code assumes that the sodium pool burning rate per unit surface area is limited by the rate of turbulent natural convective mass transport of oxygen to the

pool surface. The results produced using this assumption are consistent with previous experimental work. The limitations of SOFIRE II reflect the lack of computational resources available in the 1960s and 1970s. The containment atmosphere is modeled as a single control volume; nodalization of the walls, floor, and ceiling are limited to 3 nodes each; only convective heat transfer is modeled from the containment atmosphere to the walls and ceiling; and no heat capacity effects of the aerosols are modeled.

CONTAIN-LMR was developed by Sandia National Laboratories (SNL) from the CONTAIN 1.1 containment analysis code. CONTAIN 1.1 was developed to analyze temperature and pressure loadings and containment conditions during accident scenarios in LWRs. The sodium pool fire models from SOFIRE II and the spray fire models from NACOM [112] were incorporated into CONTAIN 1.1, resulting in CONTAIN-LMR. Modeling capabilities of CONTAIN-LMR that are not available in SOFIRE II include:

- Containment atmosphere may be divided into multiple control volumes,
- Multiple heat structures may be defined in each containment control volume,
- Permits user-defined node discretization of each heat structures,
- Utilizes thermal radiation model to simulate the heat transfer from the containment atmosphere to heat structures.

CONTAIN-LMR has several shortcomings that are currently being addressed by SNL. Most of the deficiencies stem from the discontinuation of SFR funding in the U.S. in the early 1990s and the termination of SNL support for the code. Localized, independent development of the code continued abroad in nations that were actively pursuing modern SFR designs, such as Japan and Korea, and now various versions of the code exist. All versions are now referred to as CONTAIN-LMR/1B-Mod. 1, but due to independent, unrecorded source modification, the results produced by the various versions of the CONTAIN-LMR code vary substantially. SNL is currently working to resurrect the sodium fire models from CONTAIN-LMR so that they can be incorporated into the latest version of MELCOR.

6.2 State of Knowledge Gaps

Properly assessing the release and transport of radionuclides during SFR accidents requires knowledge and modeling capabilities in a variety of areas. This section discusses a *preliminary* assessment of gaps in the state of knowledge that were encountered during the creation of this document.

Radionuclide Releases from Molten Metal Fuel

As noted in Section 4.4, one area that lacks both experimental data and theoretical models is the release of radionuclides from metal fuel during fuel melting. Determination of the release from the fuel during cladding breach is fairly straightforward and well understood, as fission gases and bond sodium will be released. The radionuclides within the fission gas can be approximated by thermodynamic equilibrium calculations, as was done for the ALMR source term discussed in Section 4.3.6. However, it is less apparent what radionuclides are released into the primary sodium when metal fuel is in the molten state in sodium, especially when at higher burnup values.

During the Fermi 1 and EBR-II fuel capsule accidents, where significant fuel melting occurred, a variety of radionuclides were released to the primary sodium. While it does not appear that many actinides (U/Pu) are released from the fuel, halogens, alkali metals, alkaline earths, and noble metals were found in the primary sodium (and for the most part, remained there). There does not appear to be clear experimental evidence regarding the approximate quantities of these materials that are released into the primary sodium during fuel melting. Almost all past fuel melting experimentation has centered on oxide fuel, since it can react with sodium. Unfortunately, these data are not translatable to metal fuel. While metal fuel is compatible with sodium, that does not preclude the release of some of its radionuclide inventory during melting.

Additional data on metal fuel melt releases may exist. As shown in Section 4.2, as part of reprocessing research in the 1960s, metal fuel elements were melted at high temperatures to gauge the release of various radionuclides. However, the use of an oxide crucible as the fuel vessel invalidates the release results for some radionuclides, since they form oxides that would not exist in the actual accident scenario. There may be additional data similar to these tests in reprocessing databases or fuel databases. Also, tens of thousands of metal fuel pins were irradiated in past U.S. SFRs, with many more heated in furnaces or subjected to testing in facilities like TREAT. Examination of these fuel elements may be possible, depending on their current status.

Table 6-2: Overview of Gap – Radionuclide Releases from Molten Metal Fuel

Gap	Possible Licensing Impediment?	Closure Suggestion
Radionuclide Release Fractions from Molten Metal Fuel in Sodium	Using available knowledge from past SFR accidents and experiments, a bounding estimate for release fractions from molten metal fuel can be made. While this likely introduces a conservative assumption, it does not preclude the satisfaction of regulatory requirements, as high retention rates of radionuclides in the sodium and primary system are likely.	**Short Term**: A thorough review should be conducted of reprocessing and fuel databases for available radionuclide release information. **Long Term**: If sufficient data cannot be found in the existing database, metal fuel melt experiments may be necessary.

Lanthanide/Actinide Data

As mentioned in Section 4.4 and in the ALMR source term analysis [55], thermodynamic data regarding the lanthanides and actinides are lacking when compared to the other radionuclide groups. In turn, thermodynamic equilibrium calculations, which can provide bounding analyses, are difficult to perform. It is questionable how valuable these data may be, as these radionuclide groups are usually not those of most concern for the majority of accident scenarios. Even if released, low solubility in sodium would likely result in high rates of adsorption and mechanical deposition for both radionuclide groups. However, additional thermodynamic knowledge may prove otherwise, as the ALMR source term calculations found a surprisingly large release fraction for europium (a lanthanide) [55].

Table 6-3: Overview of Gap – Lanthanide/Actinide Data

Gap	Possible Licensing Impediment?	Closure Suggestion
Lack of Lanthanide and Actinide Thermodynamic Data	Historically, the lanthanides and actinides are not the radionuclide groups of the highest importance for source term assessments. Most elements within the two categories would likely adsorb or mechanically deposit on structure in the primary sodium if released from the fuel. However, some elements, like europium, may not demonstrate similar behavior.	**Short Term**: A review should be conducted of recent thermodynamic data collection efforts for any potential new data. **Long Term**: If sufficient data cannot be found in the existing database, thermodynamic and solubility experiments may be necessary.

Modeling Capability for in Sodium/Cover Gas Radionuclide Behavior

Currently, the U.S. has no direct modeling capability related to radionuclide behavior during the release from the fuel pin, into the primary sodium, and into the cover gas region. Internationally, codes that address part of this process have been created, like TRACER, but no comprehensive capability exists. This deficiency exists, in part, due to the difficulty in modeling radionuclide behavior during these processes.

For example, as discussed in Section 6.1.1 and Appendix A, thermodynamic equilibrium calculations can be used to analyze the behavior of radionuclides in the primary sodium. However, homogenous mixing and equilibrium behavior are significant assumptions that highly influence the results of the calculation. In an actual accident, radionuclides may not mix homogeneously, as has been seen with NaI in experimentation, and adsorption and mechanical deposition may significantly reduce the presence of other radionuclides. Trying to accurately track the release and movement of radionuclides in the primary sodium would be nearly impossible, as computational fluid dynamics and chemical thermodynamic calculations would need to be completed simultaneously. Instead, analyses that account for the phenomena but use bounding or equilibrium values (using existing capabilities) would likely be sufficient for the level of source term analysis expected by the regulator.

If a prospective applicant were to attempt a source term calculation using currently available analytical tools, they would likely use thermodynamic equilibrium calculations or gas-liquid phase equilibrium constants (discussed in Appendix A) to determine release fractions. When taken together with experimental and accident experience, bounding values can be derived for

almost all of the radionuclides of interest. However, the one current gap that could make this process difficult is the fact that no consolidated database of information exists regarding radionuclide transport. Thermodynamic information, phase change diagrams, and gas-liquid equilibrium constants exist for most radionuclides, but finding these data (particularly the most recent or accepted data) can be difficult.

Table 6-4: Overview of Gap – Modeling Capabilities for in Sodium/Cover Gas Radionuclide Behavior

Gap	Possible Licensing Impediment?	Closure Suggestion
Modeling Capabilities for in Sodium/Cover Gas Radionuclide Behavior	A single, comprehensive tool to assess radionuclide transport in the primary sodium and cover gas is unlikely to be developed due to the complex nature of the phenomena that occur. However, available tools, such as thermodynamic equilibrium codes, coupled with experimental and accident results may provide sufficient confidence in estimates of release fractions.	A trial MST calculation should be conducted to gauge the performance and adequacy of the existing database and code suites.

Data Quality for Licensing

Since no vendor or agency has completed the NRC licensing process utilizing an MST, there is an open question regarding the quality of the data necessary for an MST analysis. It is important to recognize that all licensing, design, and fabrication efforts in a nuclear power installation require varying levels of quality assurance (QA). Structures, systems, components (SSC) and supporting qualification data must adhere to a specific standard or regulation, where the stringency of the requirement is dependent on the intended purpose of the SSC. In the case of equipment required to perform a safety function, all qualification must adhere to the NQA-1 standard set forth by the ASME [113]. Given the safety significance of an MST and any associated transport barriers, it is important to determine the quality of data required for any supporting analyses or experimentation so that the appropriate QA protocols may be followed.

For example, in SECY-93-092 [3], the NRC stated,

> Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.

Similar statements can be found in SECY-02-0139 [114] and SECY-03-0047 [115]. However, there is no specification of whether the data utilized for the MST analysis would need to meet qualification requirements, such as those regarding NQA-1 certification. A brief review of past NRC statements and accident analysis philosophy appears to show that stringent requirements may not be necessary, under certain conditions.

As described in Section 2.1, the NRC stated in SECY-93-092 that an MST "is developed using best-estimate phenomenological models…" [3]. However, in SECY-03-0047, the NRC staff recommended:

For the purpose of siting and containment/confinement decisions, the staff recommends that conservative source terms for AOOs and DBEs be used. For emergency planning purposes a best estimate source term would be acceptable.

This recommendation was approved by the Commission in the SRM to SECY-03-0047. However, in SECY-05-0006 [116], which outlines the staff's recommended approach for scenario-specific source term calculations as part of new plant licensing, a slightly different source term approach is proposed, with the key features below:

- Scenarios are to be selected from a design-specific PRA.

- Source term calculations are based on verified analytical tools.

- Source terms for compliance should be 95% confidence level values based on best-estimate calculations.

- Source terms for emergency preparedness should be mean values based on best-estimate calculations.

- Source terms for licensing decisions should reflect scenario-specific timing, form, and magnitude of the release.

While SECY-05-0006 has not been approved by the commission, it recommends the use of best-estimate calculations for all categories of events, but suggests the use of different statistical values of the resulting distribution. This is different than the broader conservative source term statement for AOOs and DBEs in SECY-03-0047. Also, SECY-05-0006 includes the caveat of "verified analytical tools" for the source term calculation. Recently, in its response to the NGNP MST white paper [11], the NRC stated [14] that the NGNP MHTGR MST definition is consistent with the original MST description in SECY-93-092, which references best-estimate phenomenological models.

As described in Section 4, for a metal-fuel pool-type SFR, it is unlikely that any accidents within the design basis will result in fuel damage. In turn, MST calculations depicting the transport of radionuclides to the environment will likely center on beyond design basis events, which have historically been analyzed using best-estimate models. While qualified fuel performance data is likely necessary for accident scenario analysis, the radionuclide behavior and phenomena data needed for MST development of accidents beyond the design basis may not need such stringent qualification.

A possible path forward to clarify regulatory requirements would entail preparing a white paper that outlines the issue and establishes a position (such as proposing the use of best-estimate models and all available phenomenological data if it can be shown that core damage accidents are unlikely to occur within the design basis accident space). While the NRC is unlikely to approve or reject such a position paper, initial review and comment, such as those received during NGNP/NRC pre-licensing discussions, can assist in developing an MST strategy and gap closure program moving forward.

Table 6-5: Overview of Gap – Data Qualification Requirements for Licensing

Gap	Possible Licensing Impediment?	Closure Suggestion
Uncertain MST Data Qualification Requirements for Licensing	There is uncertainty regarding the extent to which past experimental data related to radionuclide release and transport would need to be relied upon to form the technical basis for MST calculations supporting future designs and NRC license applications, particularly if all core damage scenarios occur outside of the design basis.	Upon further research, a white paper outlining the issue and proposed approach should be prepared and presented to the NRC for initial review and comment.

7 Summary

While SFR source term phenomena are complex, there exists the potential for significant retention of radionuclides that are released during core damage accidents. Any MST calculation seeking an accurate assessment of radionuclide release to the environment must attempt to understand and credit these phenomena. Currently, modeling and predicative capabilities do not exist for all portions of the transport process, but significant experimental experience, coupled with accident and test history, may provide sufficient insight for most phenomena. Leveraging this information, an estimate of radionuclide release for most accidents that accounts for the various retention and transport phenomena may be developed, where bounding release fraction values may be utilized if necessary.

For SFR core damage accidents, there are usually five barriers to release (fuel, clad, sodium, primary system boundary, and containment), with significant retention of radionuclides possible. Past accidents and experiments demonstrate that for the most likely core damage accidents, only the noble gases will be released from the sodium pool in significant quantities. All other radionuclides appear to have very low release fractions ($<10^{-3}$), due to high retention rates in the fuel and primary sodium. These retention mechanisms may result in the magnitude of release from a core damage accident falling below the release levels for ex-core SFR accidents.

The possible mechanistic source terms from other SFR accidents (primary sodium pool fire, cleanup system accidents, and gas-cooled moving cask accidents) are also likely to be very low. The only exception may be a sodium pot moving cask accident that results in a sodium fire, but a more detailed analysis would need to be completed by a vendor pursuing this option to fully understand the possible repercussions.

Gaps in the knowledge state do exist, especially related to radionuclide release fractions from molten metal fuel. That does not necessarily preclude an assessment of radionuclide release, as past accidents at Fermi 1 and EBR-II provide insight into failed fuel behavior; however, it should be noted that this experience is generally limited to low burnup fuel. Additional data regarding metal fuel radionuclide release may exist, and efforts should be made to investigate reprocessing and fuel performance databases. Thermodynamic data are also lacking for some of the radionuclide groups (lanthanides and actinides), but data are available for the radionuclides that are historically the most important for source term development.

In general, lessons learned during the CRBR licensing process, which focused on large energetic accidents and releases, have led to U.S. SFR design advancements that rely on inherent and passive safety mechanisms to limit core damage and severe accidents. In turn, for credible accident sequences, the thermo-mechanical and chemical benefits of metal fuel, such as its compatibility with sodium, along with the chemical properties of primary sodium, which has high radionuclide retention rates, generally limit the radionuclide release to the environment. Together, these factors mark a progression toward greater in-fuel, in-vessel radionuclide retention, which demonstrates a distinct advantage over reliance on subsequent barriers in the defense-in-depth philosophy. This evolution may facilitate future licensing discussions with regard to source term requirements, and should be reflected in the development of an SFR MST.

Bibliography

[1] Code of Federal Regulations: 10 § 50.67 , "Accident Source Term," Amended 1999.

[2] Code of Federal Regulations: 10 § 50.2, "Definitions," Amended 2014.

[3] U.S. Nuclear Regulatory Commission, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," SECY-93-092, 1993.

[4] U.S. Atomic Energy Commission, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, 1962.

[5] U.S. Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, 1995.

[6] U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, 1990.

[7] U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, 2000.

[8] U.S. Nuclear Regulatory Commission, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," SECY-03-0047, 2003.

[9] U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum for James M. Taylor, Executive Director for Operations, from Samuel J. Chilk, Secretary, dated July 30, 1993, Subject: SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," 1993.

[10] "Next Generation Nuclear Plant Licensing Strategy: A Report to Congress," 2008.

[11] Idaho National Laboratory, "Mechanistic Source Term White Paper," INL/EXT-10-17997, 2010.

[12] Idaho National Laboratory, "NGNP Fuel Qualification White Paper," INL/EXT-10-17686, 2010.

[13] U.S. Nuclear Regulatory Commission, "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms (Revision 1) - Next Generation Nuclear Plant," ML14174A845, 2014.

[14] U.S. Nuclear Regulatory Commission, "Summary Feedback on Four Key Licensing Issues, Next Generation Nuclear Plant, Project 0748," 2014.

[15] U.S. Nuclear Regulatory Commission, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," NUREG-1338, 1989.

[16] ASME/ANS, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," RA-S-1.4-2013, 2013.

[17] Code of Federal Regulations: 10 § 100, "Reactor Site Criteria," Amended 2013.

[18] U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to Operation of Fast Flux Test Facility," NUREG-0358, 1979.

[19] Project Management Corporation, "Clinch River Breeder Reactor Proejct - Preliminary Safety Analysis Report," 1976.

[20] U.S. Nuclear Regulatory Commission, "Safety Evaluation Report related to the

Construction of the Clinch River Breeder Reactor Plant," NUREG-0968, 1983.

[21] Wigeland R. and J. Cahalan, "Fast Reactor Fuel Type and Reactor Safety Performance," in *Global 2009*, Paris, 2009.

[22] U.S. Nuclear Regulatory Commission, "Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor," NUREG-1369, 1991.

[23] U.S. Nuclear Regulatory Commission, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NUREG-1368, 1994.

[24] General Electric, "PRISM Preliminary Safety Information Document," GEFR-00793, 1987.

[25] U.S. Nuclear Regulatory Commission, "Report to Congress: Advanced Reactor Licensing," 2012.

[26] Argonne National Laboratory, "Argonne, KAERI to Develop Prototype Nuclear Reactor," [Press Release], http://www.anl.gov/articles/argonne-kaeri-develop-prototype-nuclear-reactor, 2014.

[27] Chang, Y., "Technical Rationale for Metal Fuel in Fast Reactors," *Nuclear Engineering and Technology*, vol. 39, no. 3, pp. 161-170, 2007.

[28] Walter, C., Golden, G., and Olson, N, "U-Pu-Zr Metal Alloy: A Potential Fuel for LMFBRs," ANL-76-28, 1975.

[29] Walters, L. et al., "Sodium Fast Reactor Fuels and Materials: Research Needs," SAND2011-6546, 2011.

[30] Crawford, D., Porter, D., and Hayes, S., "Fuels for Sodium-cooled Fast Reactors: US Perspective," *Journal of Nuclear Materials*, vol. 371, pp. 202-231, 2007.

[31] Toshiba Corporation Power Systems Company, "Overview of Fast Reactor Development of Toshiba - 4S and TRU Burner," Presented at Nordic-Gen4 Seminar, 2014.

[32] Planchon, H., Golden, G., and Sackett, J., "Demonstration of Passive Safety Features in EBR-II," CONF-870415--3, 1987.

[33] Einzinger, R. and Seidel, B., "Irradiation Performance of Metallic Driver Fuel in Experimental Breeder Reactor II to High Burnup," *Nuclear Technology*, vol. 50, no. 1, pp. 25-39, 1980.

[34] Walters, L. and Kittel, J., "Development and Performance of Metal Fuel Elements for Fast Breeder Reactors," *Nuclear Technology*, vol. 48, no. 3, pp. 273-280, 19080.

[35] Hofman, G., Walters, L., and Bauer, T., "Metallic Fast Reactor Fuels," *Progress in Nuclear Energy*, vol. 31, no. 1/2, pp. 83-110, 1997.

[36] Denman, M., "Probabilistic Transient Analysis of Fuel Choices for Sodium Fast Reactors," *Dissertation*, 2012.

[37] Chang, Y., Finck, P., and Grandy, C., "Advanced Burner Test Reactor - Preconceptual Design Report," ANL-ABR-1, 2006.

[38] Atomic Power Development Associates, "Design of the Radiation Shield System for the Enrico Fermi Atomic Power Plant," APDA-203, 1967.

[39] Blanchet, D., Buiron, L., Stauff, N., Kim, T., and Taiwo, T., "AEN-WPRS Sodium Fast Reactor Core Definitions," 2011.

[40] Chikazawa, Y., Farmer, M., and Grandy, C., "Technology Gap Analysis on Sodium-Cooled

Reactor Fuel-Handling System Supporting Advanced Burner Reactor Development," *Nuclear Technology*, vol. 165, pp. 270-292, March 2009.

[41] International Atomic Energy Agency, "Fission and Corrosion Product Behaviour in Liquid Metal Fast Breeder Reactors (LMFBRs)," IAEA-TECDOC-687, 1993.

[42] Osterhout, M., "Control of Oxygen, Hydrogen, and Tritium in Sodium Systems at Experimental Breeder Reactor II," UAC-41069, 1978.

[43] Brehm, W. et al., "Corrosion and Fission Products in Primary Systems of Liquid Metal Cooled Reactors in the USA," in *IAEA/IWGFR Specialists' Meeting on Fission and Corrosion Prodcuts in Primary Systems of LMFBR's*, Karlsruhe, Germany, 1987.

[44] Koch, L., *Experimental Breeder Reactor-II (EBR-II): An Integrated Experimental Fast Reactor Nuclear Power Station.*, 2008.

[45] Monson, L., McDaniel, M., Knox, A., and Rice, R, "The EBR-II Cover-Gas Cleaunp System," in *IAEA Symposium on Design, Construction and Operating Experience of Demonstration Liquid Metal Fast Breeder Reactors*, 1978.

[46] Powers, D., Clement, B., Denning, R., Ohno, S., and Zeyen, R., "Advanced Sodium Fast Reactor Accident Source Terms: Research Needs," SAND2010-5506, 2010.

[47] Tentner, A., "Severe Accident Approach - Final Report, Evaluaton of Design Measures for Severe Accident Prevention and Consequence Mitigation," ANL-GENIV-128, 2010.

[48] Beck, W., Fousek, R., and Kittel, J, "The Irradition Behavior of High-Burnup Uranium Plutonium Alloy Prototype Fuel Elements," ANL-7388 (TID-4500), 1968.

[49] Pahl, R. and Wisner, R., "Steady-State Irradiation Testing of U-Pu-Zr Fuel to >18% Burnup," in *1990 International Fast Reactor Safety Meeting*, 1990.

[50] Thormeier, K., "Solubility of the Noble Gases in Liquid Sodium," *Nuclear Engineering and Design*, vol. 14, pp. 69-82, 1970.

[51] Castleman, A., "LMFBR Safety I, Fission-Product Behavior in Sodium," *Nuclear Safety*, vol. 11, no. 5, pp. 379-389, Sept-Oct 1970.

[52] Erdman. C., Kelly, J., Kirbiyik, M., and Reynolds, A., "Radionuclide Production, Transport, and Release from Normal Operation of Liquid-Metal-Cooled Fast Breeder Reactors," EPA-520/3-75-019, 1975.

[53] Christian, J., "Chemical Behavior of Iodine-131 During SRE Fuel Element Damage in July 1959: Response to Plantiffs Expert Witness Arujn Makhijani," Prepare for in re Boeing Litigation, 2005.

[54] Morewitz. H., "Radiological Source Terms for LMFBR CDAs: A State-of-the-Art Review," HMA-83-2, 1983.

[55] Schram, R., Cordfunke, E., and Huntelaar, M., "Source Term Calculations of the ALMR," ECN-R--95-021, 1995.

[56] Keilholtz, G. and Battle, G., "Fission Product Release and Transport in Liquid Metal Fast Breeder Reactors," ORNL-NSIC-37, 1969.

[57] Chellew, N. and Ader, M., "The Melt Refining of Irradiated Uranium: Application to EBR-II Fast Reactor Fuel. XI. Behavior of Iodine in Melt Refining," *Nuclear Science and Engineering*, vol. 9, pp. 82-86, 1961.

[58] Feurestein, H., Hooper, A., and Johnson, F., "Mechanisms of Release of Radioactive

Products into Liquid-Metal Coolants, their Transport within the Circuits and Removal from LMFBRs," *Atomic Enery Review*, vol. 17, no. 3, pp. 697-761, 1979.

[59] Krsul, J., "Review and Evaluation of Report of David A Lochbaum," 2006.

[60] Castleman, A., Tang, I., and Mackay. R., "Fission Product Behavior in Sodium Systems," in *IAEA Symposium on Alkali Metal Coolants,- Corrosion and System Operating Experience* , 1966.

[61] Clough, W., "The Partition of Iodine Between Liquid Sodium and the Gas Phase at 500C," *Journal of Nuclear Energy*, vol. 23, pp. 495-503, 1969.

[62] Sauter, H. and Shutz. W., "Aerosol and Activity Releases from a Contaminated Sodium Pool into an Inert Gas Atmosphere ," KfK 3504 , 1983.

[63] Haga, K., NIshizawa, Y., Watanabe, T., Miyahara, S., and Himeno, Y., "Equilibrium and Nonequilibrium Partition Coefficients of Volatile Fission Products Between Liquid Sodium and the Gas Phase," *Nuclear Reactor Safety*, vol. 97, pp. 177-185, February 1992.

[64] Zwetzig, G., "Survey of Fission- and Corrosion-Product Activity in Sodium- or Nak-Cooled Reactors," AI-AEC-MEMO-12790, 1969.

[65] Middleton, B., Parma, E., Oliver, T., Phillips, J., and LaChance, J., "The Development of a Realistic Source Term for Sodium-Cooled Fast Reactors: Assessment of Current Status and Future Needs," SAND2011-3404, 2011.

[66] Tang, I., Castleman, A., and Munkelwitz, H., "The Transport of Fission-Product Cesium from Sodium," in *155th Meeting of the American Chemical Society*, 1968.

[67] Hart, R. and Nelson, C., "Fission Product Retention Characteristics of Sodium at High Temperatures," NAA-SR-MEMO-8712, 1964.

[68] Nishimura, M., Nakagiri, T., and Miyahara, S., "Evaporation Release Behavior of Volatile Fission Products (Iodine, Cesium, and Tellurium) from Liquid Sodium Pool to the Inert Cover Gas," *O-arai Engineering Center*, 1996.

[69] Koch, M., Brockmeier, U., Schutz, W., and Unger, H., "A Code for the Prediction of Sodium and Volatile Fission Product Release from a Liquid Pool into An Inert Gas Atmosphere," *Journal of Aerosol Science*, vol. 22, no. 1, pp. S709-S712, 1991.

[70] Guon, J., "Deposition of Cesium and Barium in a Sodium-Stainless Steel System," AI-AEC-12952, 1970.

[71] Chellew, N. and Bennett, G., "The Melt Refining of Irradiated Uranium: Application to EBR-II Fast Reactor Fuel. XII. The behavior of Ruthenium, Molybdenum, Palladium, Rhodium, Technetium, Antimony, Cadmium, and Tellurium," *Nuclear Science and Engineering*, vol. 9, pp. 87-90, 1961.

[72] Okamoto, H., "Ru-U (Ruthenium-Uranium)," *Journal of Phase Equilibria and Diffusion*, vol. 33, no. 6, p. 499, 2012.

[73] Chellew, N., Bennett, G., and Trice, V., "The Melt Refining of Irradiated Uranium: Applications to EBR-II Fast Reactor Fuel: VII. The Behavior of Rare Earths, Yttrium, Barium, Strontium, and Cesium," *Nuclear Science and Engineering*, vol. 9, pp. 64-70, 1961.

[74] Predel, B., "Ce-U (Cerium-Uranium)," in *The Landolt-Börnstein Database - Group IV Physical Chemistry, Numerical Data and Functional Relationships in Science and Technology*., vol. 5.

[75] Predel, B., "U-Y (Uranium-Yttrium)," in *The Landolt-Börnstein Database - Group IV Physical Chemistry, Numerical Data and Functional Relationships in Science and Technology.*, vol. 5.

[76] Kunkel, W. et al., "LMFBR Fission Product Contamination and Control Studies - A Status Report," AI-AEC-12687, 1968.

[77] Caputi, R. and Adamson, M., "Experimental Determination of Plutonia and Urania Solubility in Liquid Metal," GEFR-SP-202, 1980.

[78] Douglas, T., "A Cryoscopic Study of the Solubility of Uranium in Liquid Sodium at 97.8 C," *Journal of Research of the National Bureau of Standards*, vol. 52, no. 5, pp. 223-226, 1954.

[79] Kurata, M. et al., "Development of Transmutation Technology of Long-Lived Nuclides: Properties of the Alloy including Minor Actnides and Rare Earths," *Central Research Institute of Electric Power Industry (CRIEPI) Research Report*, 1992.

[80] Atomics International, "SRE Fuel Element Damage," NAA-SR-4488, 1959.

[81] Hart, R., "Distribution of Fission Product Contamination in the SRE," Atomics International, NAA-SR-6890, 1962.

[82] Daniel, J., "Investigation of Releases from Santa Susana Sodium Reactor Experiment," TDR-DA/0502, 2005.

[83] Power Reactor Development Company, "Report on the Fuel Melting Incident in the Enrico Fermi Atomic Power Plant on October 5, 1966," APDA-233, 1968.

[84] Friedland, A., "Thermal-Hydraulic Analysis of the October 5, 1969 Fuel Metling Incident in the Enrico Fermi Reactor," APDA-LA-5, 1969.

[85] Atomic Power Development Associates Inc., "Compilation of Current Technical Experience at Enrico Fermi Atomic Power Plant: Monthly Report No. 3 - October 1966," APDA-CFE-3, 1967.

[86] Smith, R., Ebersole, E., Fryer, R., and Henault, P., "Fission-Product Release from an Encapsulated U-Pu-Zr Ternary Alloy," *Transactions of the American Nuclear Society*, vol. 12, no. 1, p. 180, June 1969.

[87] Smith, R., Ebersole, E., Fryer, R., and Henault, P., "Origin of Fission-Product Release in EBR-II November 23, 1967, to May 6, 1968," ANL-7604, 1970.

[88] Batte. G., Pahl, R., and Hoffman, G., "Current Status of the Run-Beyond-Cladding Breach (RBCB) Tests," ANL/FP/PP--68403, 1993.

[89] Bauer, T., Wright, A., Robinson, W., Holland, J., and Rhodes, E., "Behavior of Modern Metallic Fuel in TREAT Transient Overpower Tests," *Nuclear Fuel Cycles*, vol. 92, pp. 325-352, December 1990.

[90] Newman, R., "The Ignition and Burning Behavior of Sodium Metal in Air," *Progress in Nuclear Energy*, vol. 12, no. 2, pp. 119-147, 1983.

[91] Newman, R. and Payne, J., "Fundamental Studies of the Mechanism of Sodium Combustion," in *International Conference on Liquid Metal Technology in Energy Production*, Champion, PA, 1976.

[92] Newman, R. and Payne, J., "The Burning Rates of Sodium Pool Fires," *Combustion and Flame*, vol. 33, pp. 291-297, 1978.

[93] Newman, R., "The Role of Carbon Dioxide in the Combustion of Sodium in Air," in *LMFBR Safety Topical Meeting*, Lyon-Ecully, France, 1982.

[94] Sahoo, P., Kannan, S., Muralidharan, P., and Chandran, K., "Studies on Release and Deposition Behavior of Cesium from Contaminated Sodium Pools and Cesium Trap Development for FBTR," *Technical Committee Meeting on Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors*, 1996.

[95] Kahawara, S., Murata, T., and Sagawa, N., "Release of Sodium and Radioisotopes into Air associated with Sodium Combustion," *Journal of Nuclear Science and Technology*, vol. 13, pp. 382-388, July 1976.

[96] Berlin, M., de Montiagnac, E., Dugresne, J., and Geisse, G., "Evaluation of the Sodium Retention Factors for Fission Products and Fuel," in *Conference Internationale Sur La Surete des Reacterus Rapides a Metal Liquide et ses Aspects Conception et Fonctionnment*, Lyon, France, 1982.

[97] Deukkwang, A., Sunderland, P., and Lathrop, D., "Suppression of Sodium Fires with Liquid Nitrogen," *Fire Safety Journal*, vol. 58, pp. 204-207, 2013.

[98] U.S. Department of Energy, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," DOE-HDBK-3010-94, 1994 (reaffirmed 2013).

[99] Oak Ridge National Laboratory, "A User's Manual for the ORIGEN2 Computer Code," ORNL/TM-7175, 1980.

[100] Outotec, "HSC Chemistry 8 User's Guide," 2014.

[101] Yacout, A. and Billone, M., "Current Status of the LIFE Fast Reactors Fuel Performance Codes," in *FR13: International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios*, Paris, France, 2013.

[102] Editor: Fanning, T., "The SAS4A/SASSYS-1 Safety Analysis Code System: User's Guide," ANL/NE-12/4, 2012.

[103] Japan Nuclear Cycle Devlopment Institute (JNC), "In-vessel Source Term Analysis Code TRACER Version 2.3 - User's Manual," JNC TN9520, 2005.

[104] Miyagi, K. and Miyahara, S., "Development of In-Vessel Source Term Analysis Code, TRACER," in *IAEA Technical Committee Meeting on Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors*, 1996.

[105] Cho, D., Condiff, D., and Chan, S., "Role of Fuel Bubble Phenomenology in Assessment of LMFBR Source Term," in *ANS/ENS Fast Reactor Safety Meeting*, 1985.

[106] Borckmeier, U., Koch, M., Unger, H., and Schutz, W, "Volatile Fission Product and Sodium Release from Liquids," *Nuclear Engineering and Design*, vol. 148, pp. 499-507, 1994.

[107] Sandia National Laboratories, "CONTAIN LMR/1B-Mod. 1, A Computer Code for Containment Analysis of Accidents in Liquid-metal-cooled Nuclear Reactors," SAND--91-1490, 1993.

[108] Sandia National Laboratories, "MELCOR Computer Code Manuals," NUREG/CR-6119, SAND2000-2147, 2000.

[109] Sandia National Laboratories, "MAEROS User Manual," NUREG/CR-1391, SAND80-0822, 1982.

[110] Sandia National Laboratories, "WinMACCS, a MACCS2 Interface for Calculating Health

and Economic Conseqeunces from Accidental Release of Radioactive Materials into the Atmosphere - User's Guide and Reference Manual WinMACCS Version 3," 2007.

[111] Beiriger, P. et al., "SOFIRE II User Report," AI-AEC-13055, 1973.

[112] Brookhaven National Laboratory, "The NACOM Code for Analysis of Postulated Sodium Spray Fires in LMFBRS," NUREG/CR01405, BNL-NUREG-51180, 1979.

[113] American Society of Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facility Operation," NQA-1, 2012.

[114] U.S. Nuclear Regulatory Commission, "Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactor Designs," SECY-02-0139, 2002.

[115] U.S. Nuclear Regulatory Commission, "Policy Issues Related to Licensing Non-Light-water Reactor Designs," SECY-03-0047, 2003.

[116] U.S. Nuclear Regulatory Commission, "Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing," SECY-05-0006, 2005.

Appendix A: Sodium-Radionuclide Thermodynamics

To determine the fate of those radionuclides released into the primary sodium, three types of data must be known [60]:

1) The chemical state of the radionuclides in sodium
2) Their volatility and the extent they will vaporize from sodium
3) Their chemical state upon release to the cover gas

All three of these characteristics are related to chemical thermodynamics and chemical potential calculations involving the mixing of multiple substances. The following sections review some background information related to these calculations and the data needed to perform them.

A.1 Radionuclide Mixing and Bonding in Sodium

In order to determine how two substances will mix or bond, it is necessary to find the change in chemical potential, which is usually done by finding the change in Gibbs free energy, ΔG. The following explanation will focus on mixing, but many of the same formulas can be used to analyze bonding and reactions between substances, as will be described later. A negative change in Gibbs free energy implies that the two substances will mix spontaneously, while a positive change implies that energy is needed to achieve mixing. In other words, if:

$$\Delta G_{mix} < 0, \text{ mixing will occur spontaneously}$$
$$\Delta G_{mix} > 0, \text{ mixing will } not \text{ occur spontaneously } (i.e., \text{ energy is needed})$$

The change in Gibbs free energy (ΔG_{mix}) is determined by two components, the change in entropy (which is always negative as disorder increases), and the change in enthalpy, which may be positive or negative.

$$\Delta G_{mix} = \Delta H_{mix} - T\Delta S_{mix}$$

where,
ΔH_{mix} is the change in enthalpy (positive or negative)
$-T\Delta S_{mix}$ is the change in entropy (always negative)

An *ideal solution* is a homogenous mixture of two substances where no energy is absorbed or released during the dissolution process (*i.e.*, no change in enthalpy). Therefore, mixing will occur spontaneously for an ideal solution:

$$\Delta H_{mix} = 0 \text{ for an ideal solution}$$

In an ideal solution, the properties of the mixture, such as the vapor pressure, are simply a direct proportion of the vapor pressure of the components based on their concentration. However, there are no ideal solutions in reality. This is mainly due to intermolecular forces that are not accounted for in ideal solutions (*e.g.*, energy may be required to overcome the intermolecular repulsion of two substances in order for them to mix). In the case of an SFR MST, assuming an ideal mixture would likely underestimate the release of radionuclides to the cover gas, since it would assume that mixing spontaneously occurs for every radionuclide, which would maximize retention.

If the goal is to understand whether mixing between two substances will occur, as is the case for SFR MST evaluations, the deviation from ideal behavior must be determined. Measuring the change in Gibbs free energy directly is very difficult; instead, the deviation is defined by the *activity coefficient (γ)*, which is a number that specifies the magnitude of deviation from ideal behavior. There are two ways to determine the activity coefficient; by analyzing the deviation from ideal behavior of a property, such as vapor pressure, or by using solubility data and phase state knowledge.

In non-ideal solutions, the properties such as vapor pressure do not behave as direct proportions. If the vapor pressure in solution is known, it can be compared to ideal vapor pressure behavior and the activity coefficient can be found. However, vapor pressure data in sodium are not available for many of the radionuclides of interest. Therefore, the second approach, using solubility data is needed.

Through examination of the solubility data and phase state knowledge, the excess chemical potential can be found, which is directly related to the activity coefficient. References [55] and [60] provide a more detailed review of how this process can be conducted.

A similar process also holds for determining the reactions and bonding between substances. A negative change in Gibbs free energy implies that a reaction or bonding will occur spontaneously, with the magnitude of that negative change dictating what reaction is preferential. This means that data related to Gibbs free energy and activity coefficients are also important for determining the chemical form of elements in the fuel, sodium, and cover gas.

A.2 Radionuclide Vaporization from Sodium

The rate of vaporization of any radionuclide present in a sodium solution will depend upon the partial pressure of that radionuclide and the conditions of the gas space above the sodium. Determining the partial pressure of a radionuclide in sodium again depends on the activity coefficient, since it is a measure of the deviation from ideal behavior.

The tendency of a gas to escape or expand from a solution is call *fugacity* (f). At the expected pressures of a sodium fast reactor, the fugacity is proportional to the partial pressure of the substance in sodium,

$$f \cong P_i = P_i^0 \gamma_i x_i$$

where P_i is the partial pressure of substance i and can be found using the vapor pressure of the pure solute P_i^0, the activity coefficient γ_i, and the mole fraction x_i. As can be seen, this solution depends on the activity coefficient, described above, which must be found using data on either Gibbs free energy or solubility and phase state.

When reviewing literature about SFR source terms, one will often find vaporization data for a radionuclide given using the gas-liquid equilibrium constant K_d. This is simply the ratio of the mole fraction of the substance in the vapor to the mole fraction of the substance in the liquid at equilibrium. A temperature-dependent formula is usually given for K_d, and it provides an easy way to calculate a bounding value for the amount of a substance vaporized from the sodium

pool[39]. For example, if the K_d for cesium is known, and the release fraction of sodium to the cover gas is known (which can be estimated using ideal gas laws) then the release fraction of cesium from the sodium pool is simply the following:

$$Release\ Fraction_{Cs} = K_{d\,Cs} \times Release\ Fraction_{Na}$$

Calculating K_d involves fugacity and partial pressures. Through substitution, it can be shown that K_d is the ratio of the vapor pressure of the pure solute of the radionuclide to the vapor pressure of the pure sodium itself, times the activity coefficient.

$$K_d = \frac{\gamma_i P_i^0}{P_{Na}^0}$$

The formula depends on the activity coefficient, which again shows the importance of available data either on Gibbs free energy or solubility and phase state.

---

[39] The value is considered bounding because it is the gas-liquid *equilibrium* constant, where equilibrium between the vapor space and liquid space may not occur for some time, and a larger fraction of the radionuclide may still be present in the sodium liquid.

Appendix B: Biographies of Technical Reviewers

Dr. Richard S. Denning

Positions

Ohio State University, Nuclear Engineering Program
NRC Advisory Committee on Reactor Safeguards
      Committee member (2004 – 2006)
Ohio State University, Nuclear Engineering Program
      Adjunct Professor
      Program Chair (1999-2001) & (2006-2007)
DOE Advisory Committee on Nuclear Facility Safety
      Committee member (1987-1991)
Battelle Memorial Institute
      Held technical positions from Principal Research Scientist to Sr. Research Leader (1967-2009)

Qualifications

Dr. Denning is an internationally recognized expert in the fields of risk analysis, nuclear analysis, nuclear safety, and severe accident behavior of nuclear reactors. He has managed studies of the safety and risk of a variety of nuclear facilities including commercial nuclear power plants and a number of DOE's non-reactor nuclear facilities. Dr. Denning coordinated Battelle's efforts in WASH-1400, the first comprehensive examination of nuclear power plant risk. He also managed the analysis of source terms for the five reference plants in the NRC's reexamination of the risk of light water reactors, NUREG-1150. He participated as a technical expert in the areas of fission product behavior, severe accident loads and containment performance for the NUREG-1150 uncertainty analysis. He authored two chapters of the PRA Procedures Guide, NUREG/CR-2300.

He assisted the NRC in the development and oversight of its severe accident research program. He was a consultant to the TMI Special Inquiry Group. He was a member of DOE's Advisory Committee on Nuclear Facility Safety 1987-1991. From 1990-1992 he was the Director of Environment, Safety and Health for the multi-organization consortium that designed the heavy water new production reactor. From 1995 to 2007, he had responsibility for the oversight of safety hardware upgrades in DOE's program to improve the safety of former Soviet Union reactors (PNNL was the DOE lead laboratory). He was a member of the NRC's Advisory Committee on Reactor Safeguards from September 2004 to August 2006.

He chaired the Nuclear Engineering Program at The Ohio State University from July 1999 to June 2001 and from March 2006 to June 2007. Dr. Denning retired from the OSU faculty in May 2014. He continues to perform research in collaboration with his OSU colleagues through a consulting agreement. While a faculty member he performed research, advised graduate students, and taught classes in reactor safety analysis and risk. He has been a major contributor to the development of the dynamic event tree approach to the performance of probabilistic risk assessment. In collaboration with MIT, Dr. Denning developed methods of analysis of severe accident behavior in metal-fueled, sodium-cooled fast reactors on a DOE NEUP project. He recently was co-PI on two other DOE-NEUP projects, one examining the use of dynamic approaches to the analysis of proliferation in collaboration with BNL and the other examining the mechanistic treatment of plant aging processes in probabilistic risk assessment in collaboration with PNNL.

Mark Cunningham

Positions

<u>US Nuclear Regulatory Commission</u>

 Staff positions (1975-1989)

   • Risk assessment engineer
   • Senior risk assessment engineer

   Contributor: Reactor Safety Study (first NRC risk assessment); post-Three Mile Island investigations of emergency feedwater system reliability issues; NRC Severe Accident Policy Statement

   Member: NRC Special Inquiry Group on the Three Mile Island accident (1979-1980) Recipient: NRC Meritorious Service Award (1985) for development of nuclear reactor accident simulation models

 Senior Executive Service (1989-2011)

   • Branch chief, deputy division director, Division of Risk Analysis, Office of Nuclear Regulatory Research
   • Director, Division of Fuels, Engineering, and Radiological Research, Office of Nuclear Regulatory Research
   • Director, Division of Risk Analysis, Office of Nuclear Reactor Regulation

   Contributor: NRC Probabilistic Risk Analysis (PRA) Policy Statement; policy paper to NRC commissioners on PRA Policy Statement implementation options; regulatory guidance on use of PRA in nuclear reactor licensing actions; development of consensus standards on PRA; implementation of risk-informed fire protection requirements in operating nuclear reactors

   Recipient: Presidential Meritorious Rank Award (2004) for post-September 11 assessment of nuclear power plant vulnerabilities to external threats

<u>Post-retirement</u>

 Member: NRC Risk Management Task Force (chartered by former NRC Chairman Gregory Jaczko to "develop a strategic vision and options for adopting a more comprehensive, holistic, risk-informed, performance-based regulatory approach for nuclear reactors, materials, waste, fuel-cycle, and transportation"). Task force chaired by former Commissioner George Apostolakis; product published as NUREG-2150.

 Lecturer: Argonne National Laboratory/International Atomic Energy Agency training courses (describing how NRC has implemented nuclear safety regulations and risk assessment policies to complement more traditional engineering methods for ensuring nuclear reactor safety).

Risk Management Consultant: Clean Air Task Force (providing advice on how risk management approach proposed by NRC Risk Management Task Force could be used to improve the efficiency of advanced reactor licensing process).

Nuclear Engineering Division

Argonne National Laboratory
9700 South Cass Avenue, Bldg. 208
Argonne, IL 60439-4842

www.anl.gov

U.S. DEPARTMENT OF **ENERGY**

Argonne National Laboratory is a U.S. Department of Energy
laboratory managed by UChicago Argonne, LLC